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

Advanced Reactor Stakeholder Public Meeting

November 10, 2021

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 272 279 12#

Advanced Reactor Program

Time	Agenda	Speaker
10:00 – 10:15 am	Opening Remarks	NRC
10:15 – 11:15 am	The Technology-Inclusive, Risk-Informed Maximum Accident (TI-RI-MA) Approach – An Alternative to Probabilistic Risk Assessment	NRC
11:15 – 11:45 am	An Overview of NRC's Regulatory Requirements and Guidance on Counterfeit, Fraudulent, and Suspect Items	NRC
11:45 am – 12:15 pm	Advanced Reactor Content of Application Revised Chapter 11, "Organization and Human-System Considerations" and Chapter 12, "Post-Construction Inspection, Testing and Analysis Program"	NRC
12:15 – 1:00 pm	Lunch Break	All
1:00 – 1:45 pm	Accelerated Fuel Qualification (AFQ) White Paper	General Atomics
1:45 – 2:45 pm	Update on the Development of a Flexible Operator and Staffing Licensing Framework for Advanced Reactors	NRC
2:45 – 2:55 pm	Break	All
2:55 – 3:25 pm	FAQ for Physical Security Cat. II Fuel Cycle Facilities	NRC
3:25 – 3:55 pm	Fuel Qualification for Molten Salt Reactors	NRC/ORNL
3:55 – 4:00 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/details#advSumISRA

Advanced Reactor Integrated Schedule of Activities

Advanced Reactor Program - Summary of Integrated Schedule and Regulatory 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)	• EDO Concurrence Period
Federal Register Publication	Commission Review Period**
Public Comment Period	▼ ACRS SC/FC (Scheduled or Planned)
Draft Issuance of Deliverable	External Stakeholder Interactions
Final Issuance of Deliverable	↓ Public Meeting (Scheduled or Planned)

Version 11/4/21

Present Day

Strategy	Regulatory Activity	Commission Papers	Guidance	Rulemaking	NEIMA	Complete	2021 Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	2022 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																								
2	Identification and Assessment of Available Codes					x																								
	Development of Non-LWR Computer Models and Analytical Tools																													
	Code Assessment Reports Volume 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 Molten-Salt-Cooled Pebble Bed Reactor					x																								
	Reference plant model for Monolith-type Micro-Reactor																													
	Reference plant model for Gas-Cooled Pebble Bed Reactor																													
	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-Cooled Micro Reactor					x																								
	Reference SCALE/MELCOR plant model for High-Temperature Gas-Cooled Reactor					x																								
	Reference SCALE/MELCOR plant model for Molten Salt Cooled Pebble Bed Reactor					x																								
	Reference SCALE/MELCOR plant model for Sodium-Cooled Fast Reactor																													
	Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor																													
	MACCS radionuclide screening analysis																													
	MACCS near-field atmospheric transport and dispersion model assessment					x																								
	MACCS radionuclide properties on atmospheric transport and dosimetry																													
	MACCS near-field atmospheric transport and dispersion model improvement					x																								
	Code Assessment Report Volume 4 (Licensing and Siting Dose Assessments)							▼																						
	Phase 1 – Atmospheric Code Consolidation																													
	Code Assessment Report Volume 5 (Fuel Cycle Analysis)					x		▼																						
	Research plan and accomplishments in Materials, Chemistry, and Component Integrity for Adv. Rxs.					x																								
	Research on risk-informed and performance-based (RIPB) seismic design approaches and adopting seismic isolation technologies																													

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

The Technology-Inclusive, Risk-Informed Maximum Accident (TI-RI-MA) Approach – An Alternative to Probabilistic Risk Assessment

Advanced Reactor Stakeholder Meeting

Marty Stutzke
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

November 10, 2021

Outline

- Background: Quick history and definitions related to risk-informed regulation
- Uses of PRA in initial licensing
- Three potential licensing pathways
- Guidance development
- Path forward

Early (Post-TMI) Recommendations: Use Quantitative Risk Assessment

- ACRS letter[1] (May 16, 1979)
  - The ACRS believes that it is time to place the discussion of risk, nuclear and non-nuclear, on as quantitative basis as possible.
- Kemeny Report[1] (October 30, 1979) Recommendation #4:
  - The [Presidential] Commission recommends that continuing in-depth studies should be initiated on the probabilities and consequences (on-site and off-site) of nuclear power plant accidents, including the consequences of meltdown.
- Rogovin Report[1] (NUREG/CR-1250, January 1980), Recommendation #8:
  - The best way to improve the existing design review process is by relying in a major way upon quantitative risk analyses, and by emphasizing those accident sequences that contribute significantly to risk.

[1]Available from the Idaho National Laboratory Knowledge Management Library for the Three Mile Island Unit 2 Accident of 1979 at https://tmi2kml.inl.gov/HTML/Page1.html

Definitions (1 of 2)

- Risk triplet (Kaplan and Garrick[2], SRM-SECY-98-144[3]):
  - What can go wrong?
  - How likely is it?
  - What are the consequences?

$$\boxed{R = \{\langle s_i, p_i(\varphi_i), \zeta_i(x_i)\rangle\}}$$

- Risk assessment (SRM-SECY-98-144): A systematic method for addressing the risk triplet as it relates to the performance of a particular system (which may include a human component) to understand likely outcomes, sensitivities, areas of importance, system interactions and areas of uncertainty.
- Risk insights (SRM-SECY-98-144): The results and findings that come from risk assessments.
- Risk-informed approach (SRM-SECY-98-144): A philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety.

[2]Kaplan, S. and Garrick, B. J., "On the Quantitative Definition of Risk," *Risk Analysis*, Vol. 1, Issue 1, March 1981.
[3]NRC, "Staff Requirements – SECY-98-144 – White Paper on Risk-Informed and Performance-Based Regulation," February 24, 1998, ML003752593.

Definitions (2 of 2)

- Probabilistic risk assessment
  - **NRC online glossary**[4]: A systematic method for assessing three questions that the NRC uses to define "risk." These questions consider (1) what can go wrong, (2) how likely it is, and (3) what its consequences might be. These questions allow the NRC to understand likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty, which the staff can use to identify risk-significant scenarios. The NRC uses PRA to determine a numeric estimate of risk to provide insights into the strengths and weaknesses of the design and operation of a nuclear power plant.
  - **RG 1.200**[5]: An approach is considered to be a PRA when it (1) provides a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequences (e.g., core damage or a large early release) and their frequencies and (2) comprises specific technical elements in performing the quantification.
  - **Draft RG 1.247**[6]: A risk assessment approach is considered to be a PRA when it (1) provides a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequences (e.g., releases of radioactive material, radiological consequences) and their frequencies and (2) comprises specific PRA elements for quantifying risk.

[4]https://www.nrc.gov/reading-rm/basic-ref/glossary/probabilistic-risk-assessment-pra.html

[5]NRC, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Rev. 3, December 2020, ML20238B871.

[6]NRC, "Acceptability of Probabilistic risk Assessment Results for Advanced Non-Light Water Reactor Risk-Informed Activities," draft trial use RG 1.247, September 3, 2021, ML21246A216.

Uses of Risk Assessment in Initial Licensing

- Purpose: Thoroughly understand how and why risk assessment is used to support initial licensing.
  - Required uses of PRA
  - Expected uses of PRA
- Review of information sources:
  - Regulations
  - Rulemakings
  - Regulatory guides
  - Commission policy statements
  - Commission staff requirement memoranda
  - Standard review plans
  - IAEA SSR-2/1

Role of the PRA in Initial Licensing

- Traditional role
  - Consistent with previous DC and COL applications
  - Includes, but not limited to:
    - Searching for severe accident vulnerabilities (severe accident policy statement)
    - TMI requirement § 50.34(f)(1)(i), which under Part 52 requires LWR applicants to "Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."
    - Demonstrating that the QHOs are met (safety goal policy statement)
    - Using PRA in the design process (PRA policy statement)
  - Previously referred to as "PRA in a supporting role"
- Enhanced role
  - Any use of PRA beyond its traditional role
  - Includes, but not limited to:
    - Certain proposed required uses of PRA in preliminary 10 CFR Part 53 rule text (e.g., identifying licensing basis events; classifying systems, structures, and components; evaluating defense-in-depth)
    - Voluntary risk-informed applications (e.g., risk-managed technical specifications, risk-informed fire protection)
  - Previously referred to as "PRA in a leading role"

Uses of the PRA

Requirements/Uses	Part 50	Part 52	Part 53 Preliminary Rule Text
Submit description of PRA and its results	Currently none. Proposed in Part 50/52 lessons learned rulemaking (NRC-2019-0196; RIN 3150-AI66)	All applicants	All applicants
Develop, maintain, and upgrade PRA		COL holders	COL and OL holders
Required uses of PRA		Meet the TMI requirements in § 50.34(f)(1)(i) – Seek improvements in core and containment heat removal systems reliability	Use PRA to: • Search for severe accident vulnerabilities • Demonstrate that safety goals are met • Use PRA to evaluate changes to the facility described in FSAR (§ 53.1322) • Use PRA **<u>or generally accepted risk-informed approaches for systematically evaluating engineered systems</u>** to: o Identify LBEs o Evaluate DID o Classify SSCs o Support the FSP
Commission expectations (e.g., policy statements and SRMs)	• Search for severe accident vulnerabilities • Demonstrate that safety goals are met • Use PRA in design		
Voluntary uses of PRA	Voluntary risk-informed applications to establish or change the licensing basis		
Leveraged uses by the staff	• Focus the staff review • Inform the development of ITAACS, COL action items, D-RAP, etc. • Support oversight and inspections		

Traditional role of PRA
Enhanced role of PRA

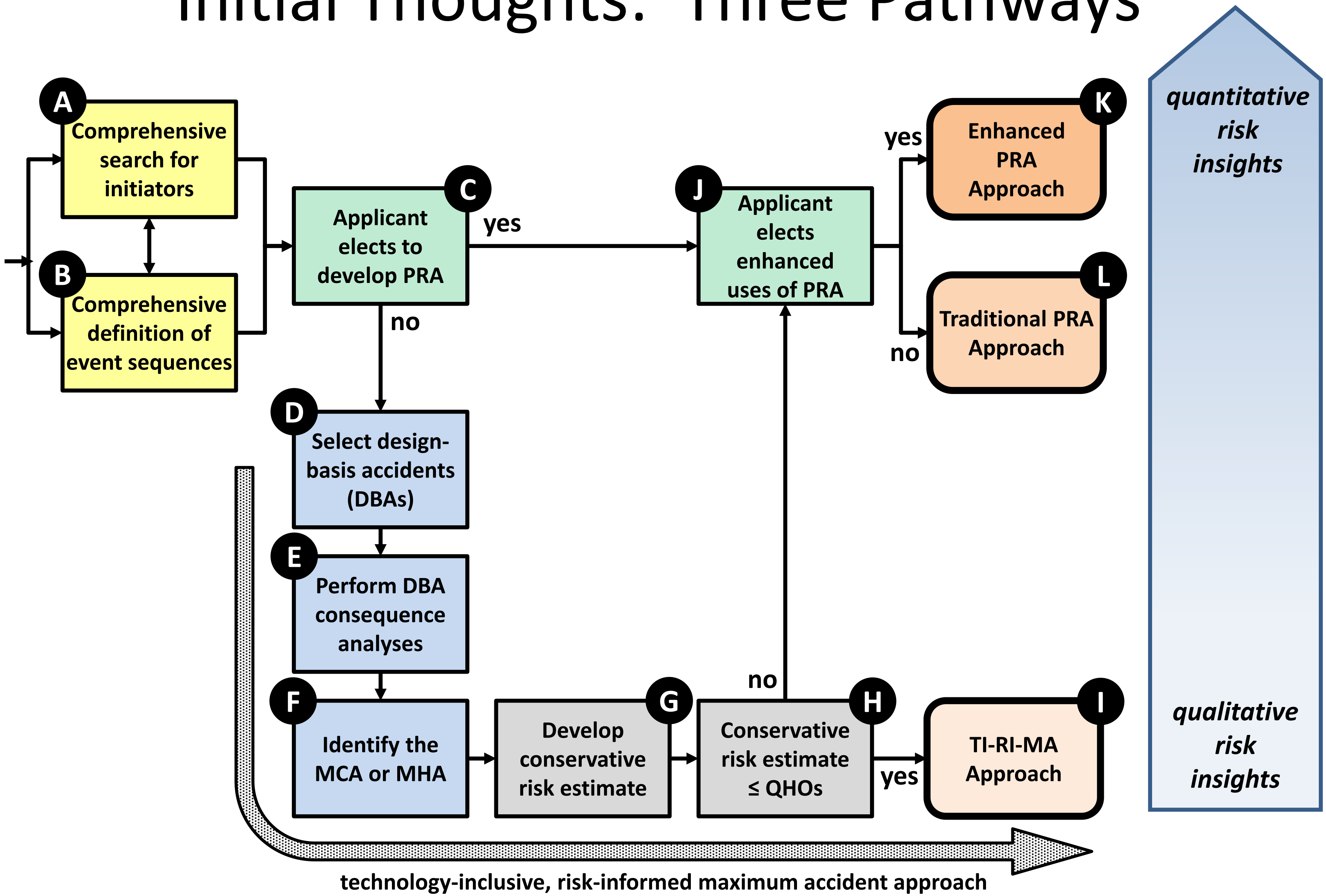
Observations

- A risk-informed approach may be based on risk insights developed from:
  - A PRA (i.e., quantitative), or
  - A qualitative risk assessment
- PRA not used to support NPUF licensing
  - Not addressed in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors"
  - Discussion with NRR/DANU staff
- Integrated safety analysis (ISAs) required by 10 CFR Part 70 for certain licensees
- The current preliminary rule text for Part 53:
  - Codifies the traditional role of the PRA
  - Adds requirements that use PRA in an enhanced role

Three potential licensing pathways

- Enhanced PRA approach
- Traditional PRA approach
- Technology-inclusive, risk-informed maximum accident (TI-RI-MA) approach

Initial Thoughts: Three Pathways



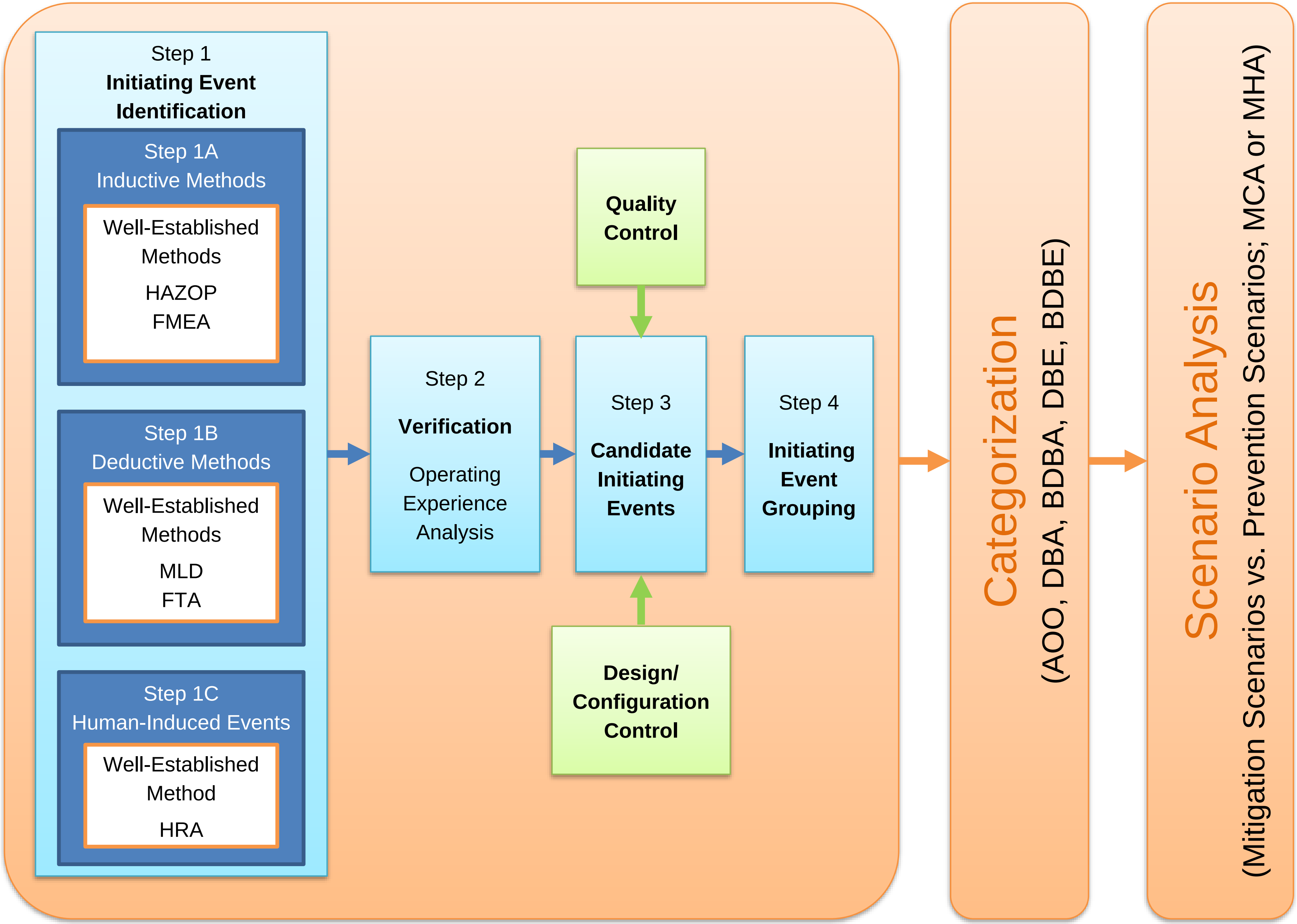
Development of "How-To" Guidance for the TI-RI-MA Approach (1 of 3)

- The staff intends to develop guidance for:
  - Box A: Comprehensive search for initiating events
  - Box B: Comprehensive definition of event sequences
  - Box D: Design-basis accident selection
  - Box E: Design-basis accident consequence analysis
  - Box F: Maximum accident (MCA or MHA) identification
  - Box G: Conservative risk estimation
- Leverage existing guidance and studies such as, but not limited to:
  - NUREG-1513, "Integrated Safety Analysis Guidance Document"
  - NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants"
  - Occupational Safety and Health Administration regulations (29 CFR 1910.119), standards, handbooks, and guidance
  - EPRI TR 3002011801, "Program on Technology Innovation: Early Integration of Safety Assessment into Advanced Reactor Design - Preliminary Body of Knowledge and Methodology"
- Guidance and preliminary rule text to be developed in parallel

Development of "How-To" Guidance for the TI-RI-MA Approach (2 of 3)

- Initial thoughts:
  - Start with a blank sheet of paper
  - Use a combination of inductive and deductive methods
  - How much searching is enough? How do you know when you are finished?
  - Focus on how plant design actually works vs. how plant design is supposed to work
  - Consolidate/group similar items
  - Be (very) careful when screening

Initiating Event and Scenario Search



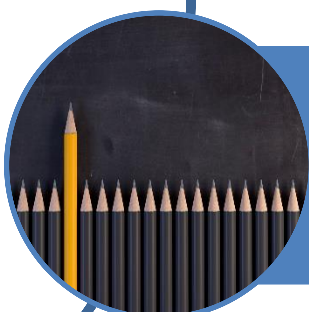
Development of "How-To" Guidance for the TI-RI-MA Approach (3 of 3)

The Advisory Committee on Reactor Safeguards, July 9, 2008.

- Multi-disciplinary team effort
- Independent review
- Documentation
  - Tell the story
  - Capture assumptions and decisions

Path Forward

NRC staff will continue to develop guidance



NRC staff will start to develop preliminary rule text

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An Overview of NRC's Regulatory Requirements and Guidance on Counterfeit, Fraudulent, and Suspect Items (CFSIs)

November 10, 2021

Deanna Zhang
NRR/DRO/IQVB

Objective

- Present an overview of the NRC’s regulatory requirements and guidance for addressing CFSIs and identify available external CFSI training.
- Engage advanced reactor vendors to raise awareness on CFSI and discuss means to prevent or mitigate CFSI in the supply chain

Background

- Concerns related to CFSI affecting NRC regulated entities prompted several NRC initiatives to support addressing CFSI concerns.
- Recent CFSI events both domestically and overseas resulted in:
  - Issuance of guidance to heighten awareness of the existing NRC regulations and how they apply to CFSI
  - Issuance of information notices on certain CFSI events
  - Creation of CFSI Technical Review Group (TRG) to evaluate events to determine whether they involve CFSI and their applicability to NRC regulated facilities
- NRC advocates a proactive approach to detect and prevent the intrusion of CFSI into SSCs intended for use as a basic component.

CFSI is encompassed in NRC regulations for quality assurance and defect reporting

Appendix B to 10 CFR Part 50 areas:

1) Design control
2) Procurement document control
3) Control of purchased materials, equipment, and services
4) Identification and control of material, parts, and components
5) Disposition of nonconforming materials, parts, or components
6) Corrective action and program effectiveness reviews

\+

10 CFR Part 21 and 10 CFR 50.55(e)

1) Evaluation of deviations and failures to comply to identify defects and failures to comply associated with substantial safety hazard
2) Notification to NRC when there is information indicating a failure to comply or a defect

Key Guidance Documents for CFSI and Recent Information Notices on CFSI events

Regulatory Issue Summary (RIS)-15-08, “Oversight of Counterfeit, Fraudulent, and Suspect Items in Nuclear Industry” heightens awareness of the existing NRC regulations and how they apply to CFSI within the scope of NRC’s regulatory jurisdiction. (ADAMS Accession No. ML15008A191)

Generic Letter 89-02, “Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products” shares information regarding elements of programs that appear to be effective in providing the capability to detect counterfeit or fraudulently marked products. https://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1989/gl89002.html

Information Notice (IN) 2018-11 Supplement 1: Kobe Steel Quality Assurance Record of Falsification
IN 2013-15 Willful Misconduct/Record Falsification and Nuclear Safety Culture
IN 2013-02 Issues Potentially Affecting Nuclear Facility Fire Safety
IN 2008-04 Counterfeit Parts Supplied to Nuclear Power Plants
NRC Bulletin 1988-010 Nonconforming Molded-Case Circuit Breakers

Other Sources of Guidance

Nuclear Industry Guidance

- EPRI Technical Report 3002002276 Plant Support Engineering: Counterfeit and Fraudulent Items – Mitigating the Increasing Risk
- NEI 14-09, Revision 1, Guidelines for Implementation of 10 CFR Part 21 Reporting of Defects and Noncompliance, as endorsed in NRC RG 1.234

International Guidance

- IAEA Publication on Managing Counterfeit and Fraudulent Items in the Nuclear Industry
- NEA MDEP CP-VICWG-04 Common Position on Counterfeit, Fraudulent, and Suspect Items Procedure

Resources from other Federal Agencies

- Department of Homeland Security National Intellectual Property Rights Coordination Center https://www.iprcenter.gov/
- Department of Energy (DOE) Operating Experience Committee https://www.energy.gov/ehss/doe-corporate-operating-experience-program

CFSI Training Offerings

- NRC has issued a list of CFSI-related training offerings as part of IN 2012-22, “Counterfeit, Fraudulent, Suspect Item (CFSI) Training Offerings,” which has been updated in 2019 (ADAMS Accession No. ML19017A117).
- NRC continues to engage stakeholders to enhance awareness on CFSI and disseminate information on CFSI-related events in the nuclear industry.

Prevention and Mitigation of CFSI in Advanced Reactor Supply Chain: Key Takeaways

- Incorporate processes to verify that products are authentic using receipt inspection, procurement controls, vendor authentication tools
- Maintain traceability of products within the supply chain and reduce risk of counterfeit products by procuring from authorized resellers mitigate
- Increase awareness of CFSI through training and coordinating with industry and government organizations involved in preventing and mitigating CFSI

Questions

Acronyms

- CFR: Code of Federal Regulations
- CFSI: counterfeit, fraudulent, and suspect item
- DHS: Department of Homeland Security
- DOE: Department of Energy
- EPRI: Electric Power Research Institute
- IAEA: International Atomic Energy Agency
- IN: Information Notice
- IPR: Intellectual Property Rights (Coordination Center)
- MDEP: Multi-national Design Evaluation Program
- NEA: Nuclear Energy Agency
- NEI: Nuclear Energy Institute
- RIS: Regulatory Information Summary
- RG: Regulatory Guide
- VICWG: Vendor Inspection Co-operation Working Group

Resources

https://www.nrc.gov/about-nrc/cfsi.html

https://www.iprcenter.gov/

https://www.epri.com/research/products/3002002276

https://www.iaea.org/publications/11182/managing-counterfeit-and-fraudulent-items-in-the-nuclear-industry

https://www.energy.gov/ehss/doe-corporate-operating-experience-program

Stakeholder's Meeting

Advanced Reactor Content of Application Project (ARCAP)

Chapter 11 "Organization and Human-System Consideration"

Interim Staff Guidance (Draft)

November 2021

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Background

- The Advanced Reactor Content of Application Project (ARCAP) is developing guidance to support the review of non-light-water reactors (non-LWRs), modular LWRs and stationary micro-reactors. This project encompasses industry-led Technology-Inclusive Content of Application Project (TICAP).

- SAR structure consists of 12 main chapters. TICAP is applicable to portions of first 8 SAR chapters. ARCAP addresses SAR Chapters 9, 10, 11, and 12.

- This guidance for SAR Chapter 11, “Organization and Human-System Considerations,” under development as a draft Interim Staff Guidance (ISG) document, currently a draft white paper (ML21309A020).

- This latest version of the guidance expands on the earlier version to provide additional proposed guidance for human factors engineering, operator licensing, operator training and staffing.

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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 Classification
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 and Human-System Considerations
12. Post-construction Inspection, Testing and Analysis 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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Updated Guidance Areas

- Included applicability section
  - This ISG will be applicable to non-LWRs, stationary micro reactors and small modular LWRs submitting applications for a construction permit (CP) or operating license (OL) under Part 50 or for a combined license (COL) under Part 52.

- Human Factors Engineering (HFE)
  - NRC staff identified need to provide guidance in this area to supplement licensing modernization project (LMP) and pending associated TICAP guidance
    - LMP provides insights but provides limited guidance on how to develop a HFE program
  - ARCAP Chapter 11 ISG covers HFE information that would support NRC findings

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Updated Guidance Areas

- Human Factors Engineering (continued)
  - References guidance found in:
    - NUREG-0711, "Human Factors Engineering Program Review Model," and
    - Scalable HFE approach being considered for Part 53 ("Final Report Development of HFE Review Guidance for Advanced Reactors," (ML21287A088)

Updated Guidance Areas

- Licensed Operator Training
  - Staff interested in stakeholder feedback on how best to capture full scope of proposed guidance
    - Staff notes that proposed ARCAP guidance includes information that is not necessary to support an operating license or combined license issuance
    - Staff's proposed guidance provides a holistic approach for operator licensing
      - Based on insights from recent LWR applications
      - Portions of guidance could be split out to other non-application guidance (e.g., guidance that supports operator licensing and inspection processes)

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Updated Guidance Areas

- Licensed Operator Training (continued)
  - Proposed guidance includes areas such as:
    - Description of how chosen examination methods, structures, and passing scores, support examination validity, reliability, and fairness
    - Procedure development used to ensure examination material: 1) writer's guide requirements are met that addresses technical specification and FSAR requirements, 2) technical review to ensure procedure is correct for proper operation of the plant
    - Description and qualification of simulator used to administer initial operator licensing examinations
    - Timeline for operator licensing examinations

Updated Guidance Areas

- Licensed Operator Training (continued)
  - Proposed guidance includes areas such as (continued):
    - Use of simulator for operation training experience and examinations during construction
    - Operator license issuance prior to fuel load
- Operator Staffing
  - Additional proposed guidance provided
    - Option of providing technical basis for control room staffing in conjunction with control room configuration that would support capturing requirements in design certification rulemaking
    - Provide technical basis that could support a future exemption from §§ 50.54(m) and/or 50.54(k) requirements

QUESTIONS?

Stakeholder’s Meeting

Advanced Reactor Content of Application Project (ARCAP)

Chapter 12 “Post-construction Inspection, Testing, and Analysis Program”

Interim Staff Guidance (Draft)

November 2021

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. This project encompasses industry-led Technology-Inclusive Content of Application Project (TICAP).

- SAR structure consists of 12 main chapters. TICAP is applicable to portions of first 8 SAR chapters. ARCAP addresses SAR Chapters 9, 10, 11, and 12.

- This guidance for SAR Chapter 12, 'Post-Construction Inspection and Analysis Program,' (PITAP) under development as a draft Interim Staff Guidance (ISG) document, currently a draft white paper (ML21294A266)."

- This latest version of the guidance expands on the earlier version to address inspections and analysis verification, including inspections, tests, analyses, and acceptance criteria (ITAAC).

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

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Background (cont.)

- This ISG will be applicable to non-LWRs, stationary micro reactors and small modular LWRs submitting applications for a construction permit (CP) or operating license (OL) under 10 CFR Part 50 or for a design certification (DC), a combined license (COL), or a manufacturing license (ML) under 10 CFR Part 52.

- This ISG differentiates between 10 CFR Part 52 applicants that must include ITAAC and 10 CFR Part 50 applications that are not required to include ITAAC.

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

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Requirements

- Post-construction inspection, testing, and analysis are required, in part, by regulations for applicants to provide a description of their quality assurance programs, as required by 10 CFR Part 50, Appendix B.
- These quality assurance requirements are also included in 10 CFR 50.34(a)(7) for CP applicants and in 10 CFR 50.34(b)(6) for OL applicants. In addition, similar COL requirements associated with quality assurance are contained in 10 CFR 52.79(a)(25). Some advanced reactor applicants for whom the requirements of 10 CFR 50.43 apply will find similar requirements contained in 10 CFR 50.43(e)(1).
- Requirements to describe preoperational testing and initial operations in OL and COL applications are contained in 50.34(b)(6)(iii) and 52.79(a)(28), respectively.

Requirements (continued)

- The following regulations require that applications contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in conformity with the license, the design certification for a facility that incorporates one, the provisions of the Act, and the Commission's rules and regulations:
  - For DC applications: 10 CFR 52.47(b)(1)
  - For COL applications: 10 CFR 52.80
  - For ML applications: 10 CFR 52.158(a)

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Requirements (continued)

- Requirements regarding Commission findings:

  The following regulations require, in part, that the Commission make a finding that the facility has been constructed and will be operated in accordance with the [design certification or license], the provisions of the Act, and the rules and regulations of the Commission.

  - For DCs: 10 CFR 52.54, “Issuance of standard design certification”
  - For OLs: 10 CFR 50.57, “Issuance of operating license”
  - For COLs: 10 CFR 52.97, “Issuance of combined licenses”

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Referenced Guidance

- The ISG references the guidance in NUREG-0800, “Standard Review Plan,” (SRP) Section 14.3, “Inspections, Tests, Analysis, and Acceptance Criteria,” with the caveat that the guidance in SRP 14.3 is applicable to LWRs but may contain insights that are useful for non-LWR application reviews. SRP 14.3 guidance in Appendix C, “Detailed Review Guidance,” may only be applicable if the features described are considered within the scope of Safety Related or safety-significant systems covered by this ISG.
- Guidance regarding PITAP and ITAAC for Emergency Planning and Physical Security Hardware are not addressed in this ISG; but the LWR guidance for these topics in SRP Sections 14.3.10 and 12, respectively, is generally technology-neutral and may be adapted for use with non-LWRs.
- Although guidance provided in Regulatory Guide (RG) 1.68, *Initial Test Programs For Water-Cooled Nuclear Power Plants*, is specific to LWRs, applicants may use this RG to gain insights that could inform the development of initial test programs for advanced reactors.

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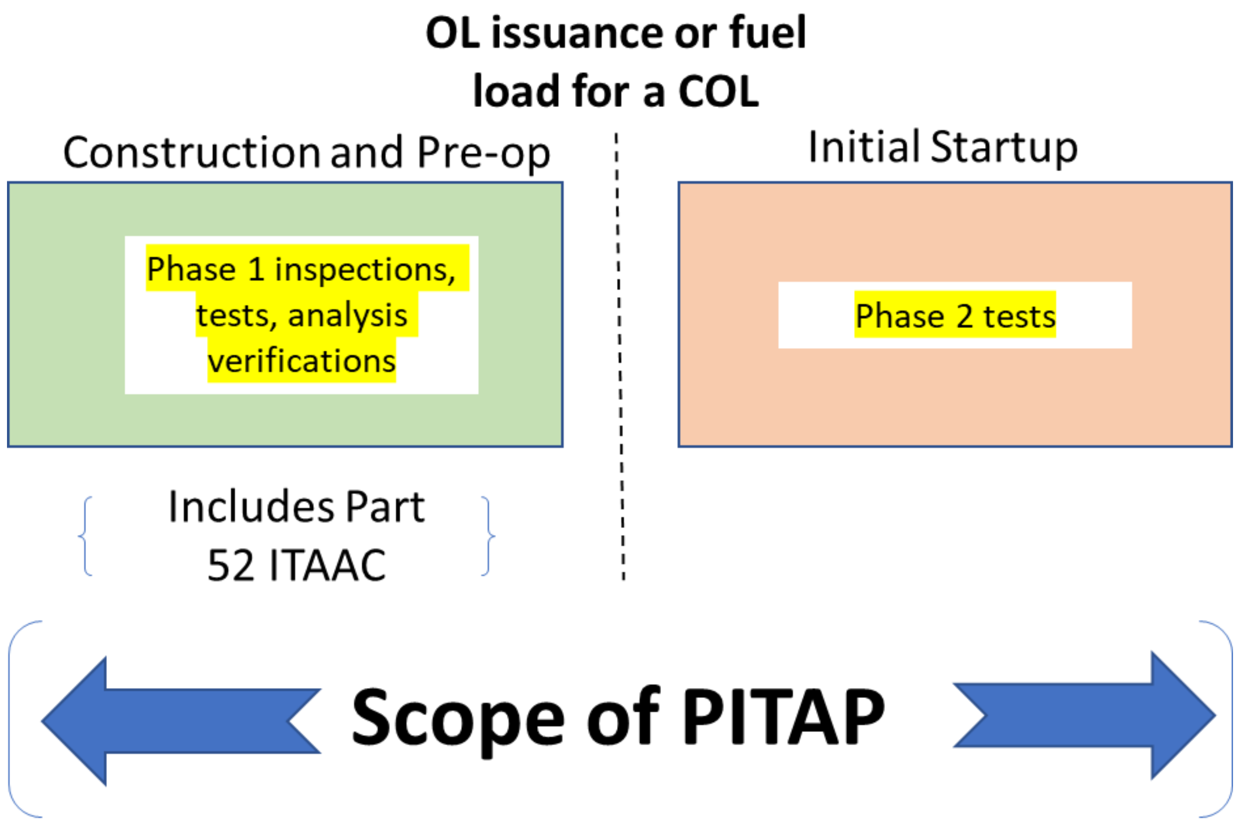
Scope

- The ISG consists of guidance related to
  - post-construction inspection, preoperational testing (i.e., tests conducted following construction and construction-related testing, but prior to initial fuel load), analysis verification, and
  - initial startup testing (i.e., tests conducted during and after initial fuel load, up to and including initial power ascension).
- This ISG is intended to provide guidance to the NRC staff regarding application content that would support making the finding that the applicant has met the referenced regulations.

Scope (continued)



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Objectives

- The primary objective of the PITAP is to demonstrate, to the extent possible, that the Safety-Related (SR) and safety-significant structures, systems and components (SSCs) have been constructed and will be operated in accordance with the design and as described in the safety analysis report.

Objectives (cont.)

- Additional objectives of the PITAP include:
  - Providing reasonable assurance that the facility exhibits the performance and associated safety margins that are described in the design;
  - Satisfying any license conditions associated with the PITAP;
  - Obtaining as-built data to validate the analytical assumptions, limits, and/or models;
  - Familiarizing the plant's operating and technical staff with operation of the facility; and
  - Verifying the adequacy of the plant operating and emergency procedures.

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Guidance Topics

Phase 1 - Preoperational Inspection, Testing, and Analysis Verification

- Inspections - The PITAP (or referenced elements of the quality assurance program) should include a post-construction (preoperational) inspection program that addresses verification of items such as:
  - Basic configuration and key design features for SR and safety-significant SSCs. This activity includes inspection of the functional arrangement of the as-built SR and safety-significant SSCs described in the safety analysis report.
  - Electrical separation for SR and safety-significant SSCs where required.
  - Materials of construction for SR and safety-significant SSCs per approved design codes and standards (e.g., ASME Code Section III, Section VIII, etc.)

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Guidance Topics

Phase 1 - Preoperational Inspection, Testing, and Analysis Verification (continued)

- Testing - The PITAP should include a post-construction (preoperational) testing program for SR and safety-significant SSCs that addresses items such as:
  - Reactivity control functions
  - Heat removal functions
  - Containment of radioactive material
  - Testing required by consensus design codes and standards applied in the design (e.g., ASME, IEEE)
  - Flow induced system vibration and thermal expansion tests
  - Electrical system performance for normal and emergency power
  - Equipment identified as necessary for defense-in-depth
  - Instrumentation and control (I&C) systems relied upon in the safety analysis to perform SR or safety-significant functions

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Guidance Topics

Phase 1 - Preoperational Inspection, Testing, and Analysis Verification (continued)

- Analysis - The PITAP (or referenced quality assurance program element) should include a description of what important analysis of SR and safety-significant SSCs should be verified including areas such as:
  - Thermal and hydraulic analysis important to the performance of required safety functions
  - Seismic analysis
  - Verification of equipment required to be qualified for a harsh environment
  - Critical assumptions from transient and accident analysis including barrier performance and effluent release calculations
  - For I&C systems, analytical limits associated with each key variable, the ranges (normal, abnormal, and accident conditions), and the rates of change for these variables

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Guidance Topics (cont.)

Phase 2 - Initial Startup Testing

- Testing - The PITAP should include a post-construction (initial startup) testing program for SR and safety-significant SSCs that addresses items such as:
  - Initial fuel loading and reactor physics tests
  - Low power testing
  - Power ascension testing
  - Performance of residual heat removal systems
  - Performance of liquid and gaseous waste systems
  - Performance of first-of-a-kind, inherent or passive safety features
  - Flow induced vibration and thermal expansion within design limits

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Guidance Topics (cont.)

General Guidelines

- Guidance for ensuring that the description of the PITAP in the application addresses programmatic items related to the development and conduct of the PITAP, such as:
  - The PITAP objectives, including the objectives of each phase of the program.
  - The scope of each phase of the PITAP.
  - The organization and responsibilities for conduct and control of the inspection and testing program.
  - A general schedule and sequence for conducting the inspections and tests, including established hold points.
  - The extent to which the test program will use plant operating, emergency and surveillance procedures and technical specifications.

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Guidance Topics (cont.)

Guidelines for Testing

- Guidance for ensuring that the application includes a general description for each test, or group of similar tests (i.e., test abstract), to be conducted.
- The focus of the test descriptions should be on providing the bases for the tests and test conditions selected, instrumentation to be used, and a description of how the tests will confirm the performance of the SSCs.
- The PITAP development should also take into consideration PITAP experience at other similar facilities and include measures to avoid problems they have had.

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Guidance Topics (cont.)

General Responsibilities

- Guidance for ensuring that the application describes the responsibilities and guidelines for conduct of the PITAP, such as:
  - Defining the qualifications of the personnel managing, conducting, and reviewing the inspection, test, and analysis verification program results.
  - Providing training as necessary to ensure that personnel are ready to perform their functions.
  - Developing the testing objectives, schedule, sequence, prerequisites, procedures safety precautions and acceptance criteria.
  - Managing, controlling, and approving key aspects (e.g., prerequisites, procedures) of the test program.
  - Establishing a plant review committee to review, evaluate, and disposition the inspection, test, and analysis verification results.

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QUESTIONS?

Advanced Reactor Stakeholder Public Meeting

Break

Meeting will resume at 1pm EST

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 272 279 12#

Accelerated Fuel Qualification (AFQ) White Paper Overview

Presented at the NRC Advanced Reactors Stakeholder Meeting

November 10, 2021

Presented by

Ron S. Faibish, PhD

General Atomics Electromagnetic Systems (GA-EMS)

The AFQ White Paper Task Force (in alphabetic order):

Framatome

General Atomics

Idaho National Laboratory

Los Alamos National Laboratory

Oak Ridge National Laboratory

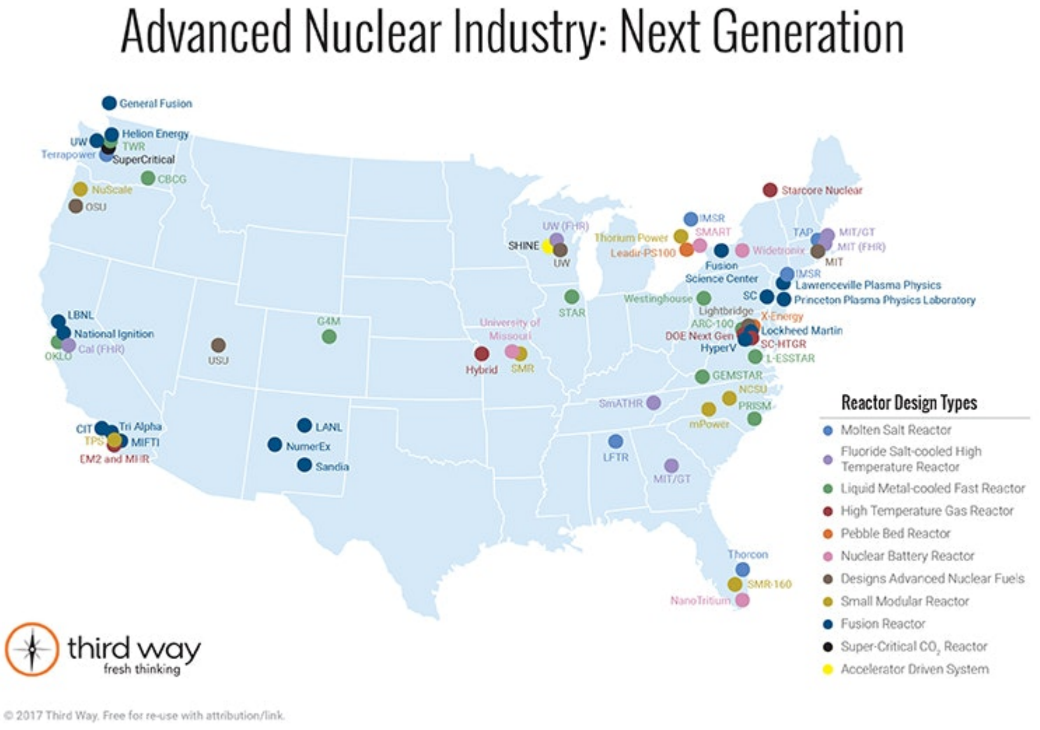
Westinghouse Electric Company

The AFQ Working Group

- The AFQ Working Group is a grassroots, industry-driven group with participants from industry, national labs, academia, DOE and NRC
- The AFQ Working Group (WG) is charged to develop a methodology that is generalized and is applicable to all fuel types
- Three key elements to the WG charge:
  - 1) adoption of physics-informed advanced nuclear fuel performance modeling and simulation (M&S) tools
  - 2) use of targeted experiments to validate the modeling, under condition that are as prototypic as possible
  - 3) use of risk-informed, performance-based decision-making tools that take into account uncertainty and provide information necessary for regulatory decisions (see https://www.nrc.gov/about-nrc/regulatory/risk-informed/concept.html)
- Main tool of implementation: information exchange workshops

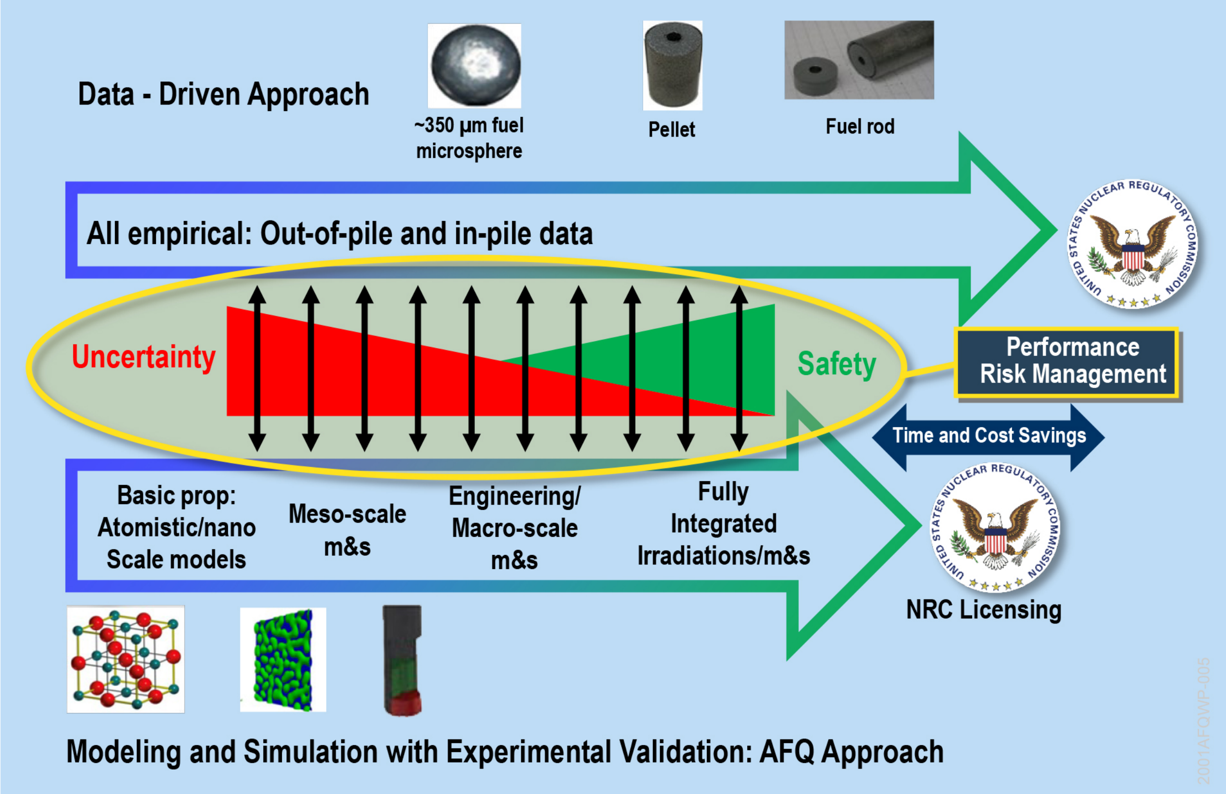
Motivation

- **Overarching motivation: Remove fuel qualification from the critical path** of advanced reactors licensing to support more timely deployment of new reactors with new fuel systems
- **Formalize the AFQ methodology** with details on enabling experimental and modeling tools and their use
- Promote the "***Adoption of the AFQ methodology by industry, and recognition of the methodology by the Nuclear Regulatory Commission (NRC)****, would facilitate more efficient and timely qualification of new fuel systems.*"



Introduction

- The AFQ methodology brings together **advanced, physics-informed nuclear fuel performance M&S with targeted experiments** to significantly reduce the time and cost to qualify new fuels
- Use of **targeted, separate-effects** and accelerated experiments **enables higher quality, fewer, more effective integral irradiation** experiments to validate and acquire the data needed to support the safety case
- **Enabling AFQ: Three main phases discussed in the paper**
  - Phase 1: Data Compilation and Physics-Based Modeling
  - Phase 2: Model Validation
  - Phase 3: Essential and Limited Integral Testing



The goal of AFQ is to reduce the time to qualify new fuels from 20 years to as few as 5 years

Key Elements of AFQ

- **High-fidelity, physics-based** modeling and simulation (M&S) tools that adequately describe the fuel performance
- **Out-of-pile and in-pile targeted experiments** that efficiently span the range of relevant parameters to provide data that may either be used to construct semi-empirical models or to efficiently validate physics and mechanistic models
- Execution of coordinated **experimental testing and M&S activities, in parallel**.
- Incorporation of **specialized and accelerated testing methods** to obtain relevant data more quickly, such as the Fission Accelerated Steady State Test (FAST) or HFIR MiniFuel irradiation methods
- The implementation of AFQ must be tailored to specific reactor type, fuel form, and associated safety case

Adoption of the AFQ methodology by industry, and its recognition by the NRC would facilitate efficient and timely qualification of new fuel systems

AFQ-Enabling Modeling & Simulation Tools

M&S are key in the AFQ methodology. Mechanistic models of fuel performance based on a multiscale methodology can be utilized to:

- Accurately interpolate between sparse experimental data on irradiated fuels
- Provide a detailed analysis of experimental results to reveal and understand governing phenomena
- Design future experiments to strategically target key unknowns or regimes.
- Potentially identify optimized fuel compositions
- Be a key aid to informing fabrication studies or manufacturing activities (although this is not the subject of this white paper)

Data produced via M&S do NOT replace integral tests and other experimental data – they are augmenting them

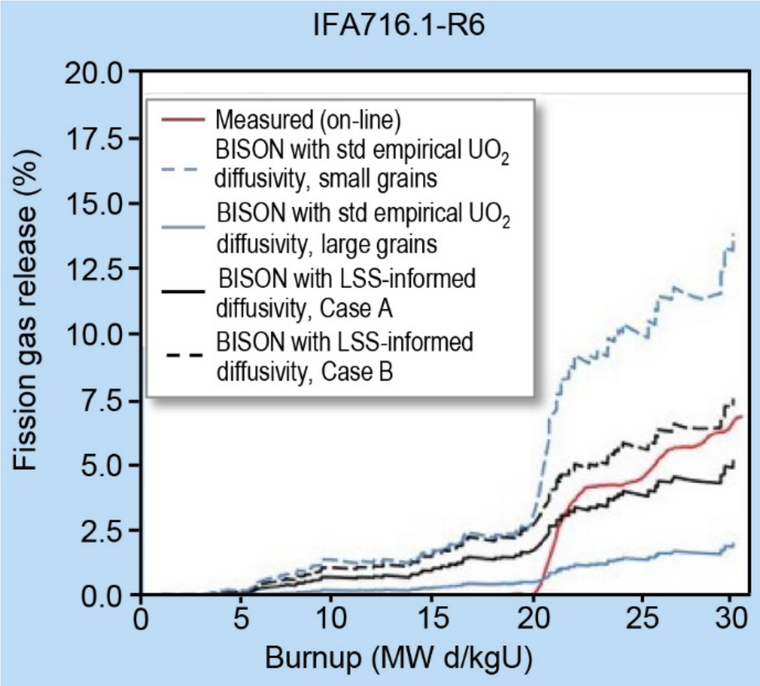
Multi-Scale Modeling & Simulation

Least Linear Scale (LLS) tools – Density functional theory (DFT), molecular dynamics (MD), dislocation dynamics (DD), cluster dynamics, phase-field simulations, crystal plasticity methods, and more, are available to be applied to nuclear fuel problems

Bison – Engineering-scale fuel performance code built on MOOSE with inherent ability to utilize modern HPC platforms, can assess multiple fuel types and complex geometries – uses both empirical materials models and multi-scale models

Data Science – scale bridging by development of reduced order models (ROMs) from mechanistic models and uncertainty quantification and sensitivity analysis – the glue that ties experiments, mechanistic and empirical models together

Example: Fission gas release in doped UO_2



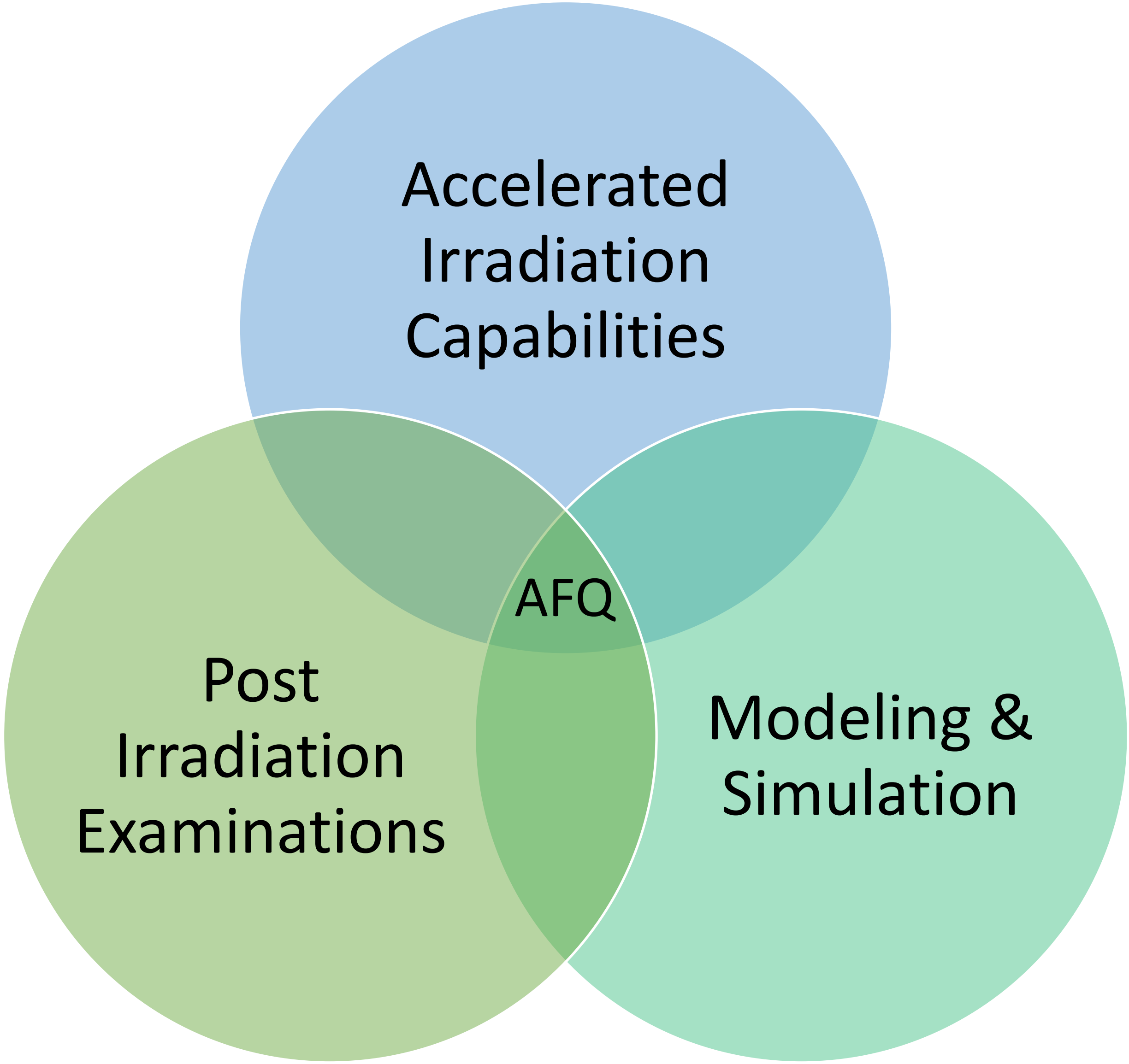
M.W.D Cooper, et al., Fission gas diffusion and release for Cr_2O_3-doped UO_2: From the atomic to the engineering scale, Journal of Nuclear Materials 545, 152590 (2021).

Mechanistic multi-scale modeling, validated by separate effects data, can provide additional data to that obtained by traditional integral tests

Advanced Experimental Tools

- Old Paradigm: **Serial and Limiting**
  - Numerous prototypic integral test campaigns
  - Develop empirical fits with limited operational regime
  - Applicability limited to data regime
    - Additional test campaigns required to broad applicability

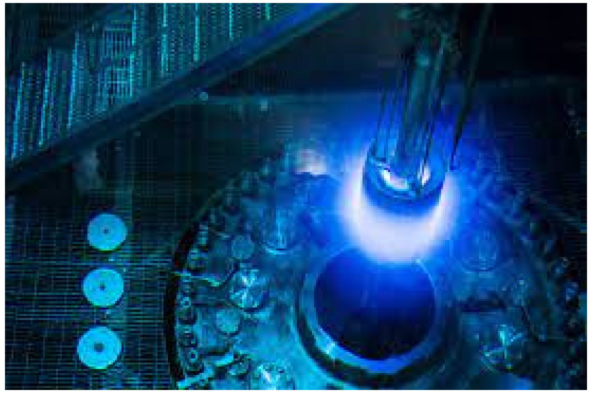
- New Paradigm: **Ties accelerated irradiation testing, advanced M&S, and high-throughput characterization at microstructural scale together**
  - Reduced integral data needs and accelerated irradiations
  - Bridge phenomenological and length scale gaps with advanced PIE
  - Develop mechanistic models bounding the application



Advanced Experimental Tools

Accelerated Experimental Capabilities

- Separate Effects: **MiniFuel**
  - Simplified design, analysis, and PIE
  - Multiple samples and conditions
  - Option to accelerate sample burnup
- Integral Irradiations: **FAST (Fission Accelerated Steady-state Testing)**
  - Burnup acceleration
    - Reduce irradiation times
  - Maintains semi-prototypic fuel conditions
- Inform/validate advanced M&S
  - Understand and assess non-prototypicalities



Advanced PIE Capabilities

- Types of Data
  - microstructure and form
  - chemistry and composition
  - thermo-mechanical properties
  - three-dimensional reconstructions
- High-throughput capabilities generate comprehensive data sets spanning length scales
- Identifies key structure-processing-properties that impact performance
  - Identifies material's strengths and weaknesses
- Used to inform the development of physics-based material models
  - Enable increasingly predictive models

The Overall Reactor System and AFQ

- For any reactor system, we need to qualify fuel such that we

  "demonstrate[e] that a fuel product fabricated in accordance with a specification behaves as assumed or described in the applicable licensing safety case, and with the reliability necessary for economic operation of the reactor plant"

- A licensing methodology - including one which utilizes AFQ - needs to:
  - Define an envelope of normal operating conditions and accident scenarios
  - Define failure modes in the above conditions
  - Define requirements to ensure that the failures will not happen, with a high degree of certainty

The Overall Reactor System and AFQ

- As stated on the prior slide, a reactor system's fuel and structural materials must be able to withstand the "fuel performance envelope" of a given design
  - AFQ can be used to accelerate feasibility assessments of fuel and reactor materials (i.e. screen materials, expedite experimental testing, etc.)
- AFQ models can be used to explore sensitivities in key parameters and to understand the influence of materials selection criteria on ultimate reactor system performance
- Novel experiments, advanced modeling, and on-line instrumentation are designed to expedite materials model development for use in codes and methods
- In new reactor designs, AFQ modeling and advanced experiments can aid in the identification and characterization of novel failure modes

Importance of the Phenomena Identification and Ranking Table (PIRT)

- **The PIRT is a systematic way of organizing information to help guide research or development of regulatory requirements**
- Once the system constraints (reactor system and operating parameters) during normal operation and postulated accident scenarios and the fuel design concepts are identified, potential failure mechanisms are considered with respect to achieving basic design functions
- The phenomena leading to the failure are then investigated and ranked

TABLE 1. MATRIX OF KNOWLEDGE VERSUS IMPACT

		Importance		
		High	Medium	Low
Knowledge Level	Known			
	Partially Known	*		
	Unknown	*	*	

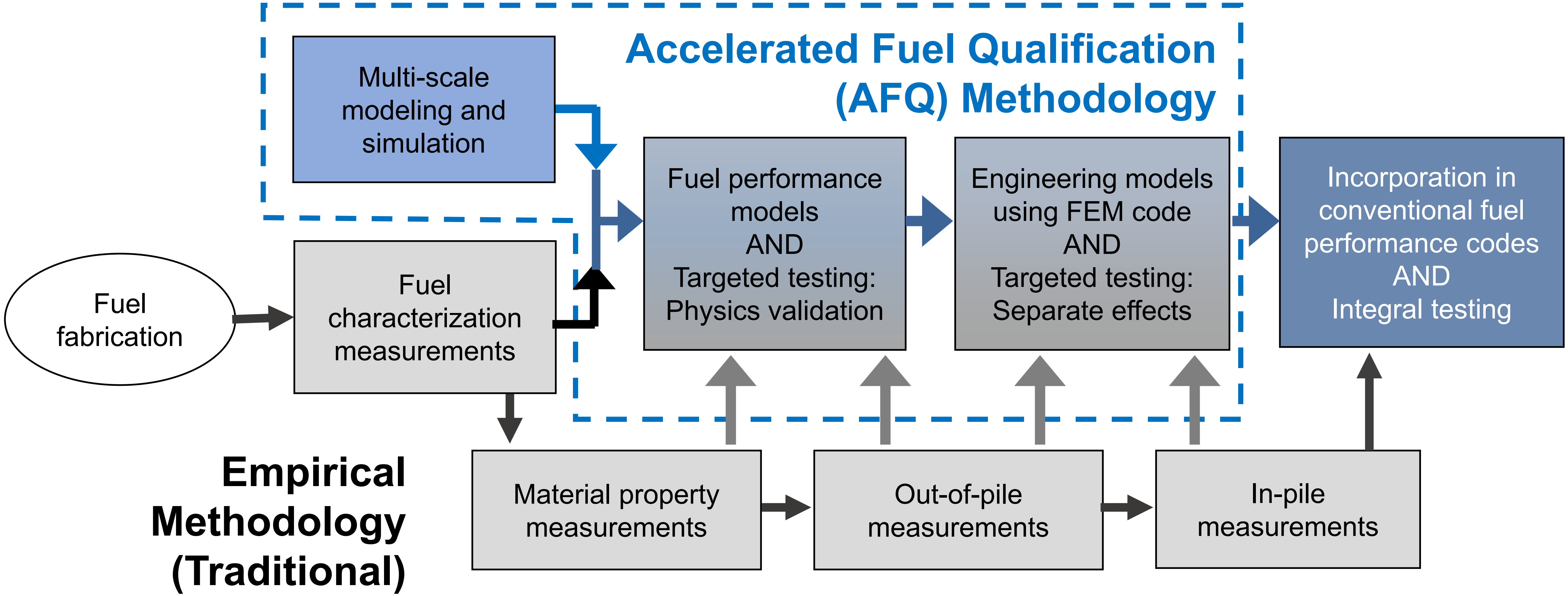
* Fuel systems falling into areas indicated by an asterisk benefit from AFQ the most

- PIRTs identify gaps in the understanding of fuel failure and damage mechanisms to focus integral irradiation testing and continued development of M&S

The PIRT Process

- A collaborative effort between fuel developers and national labs is utilized in the iterative process of ranking the phenomena to prioritize the use of engineering-scale M&S tools in combination with separate-effects testing
- The complexity of the new fuel design, or change from the currently licensed, in combination with the reactor design, determine the degree of the PIRT that needs to be completed
- A well-informed, focused and collaborative effort prioritizing R&D on lower-knowledge phenomena will contribute to a reduction in overall fuel qualification development time

Essence of AFQ: To Build on Existing Efforts and Incorporate New Tools For Efficient Demonstration and Adoption of “New” Fuels

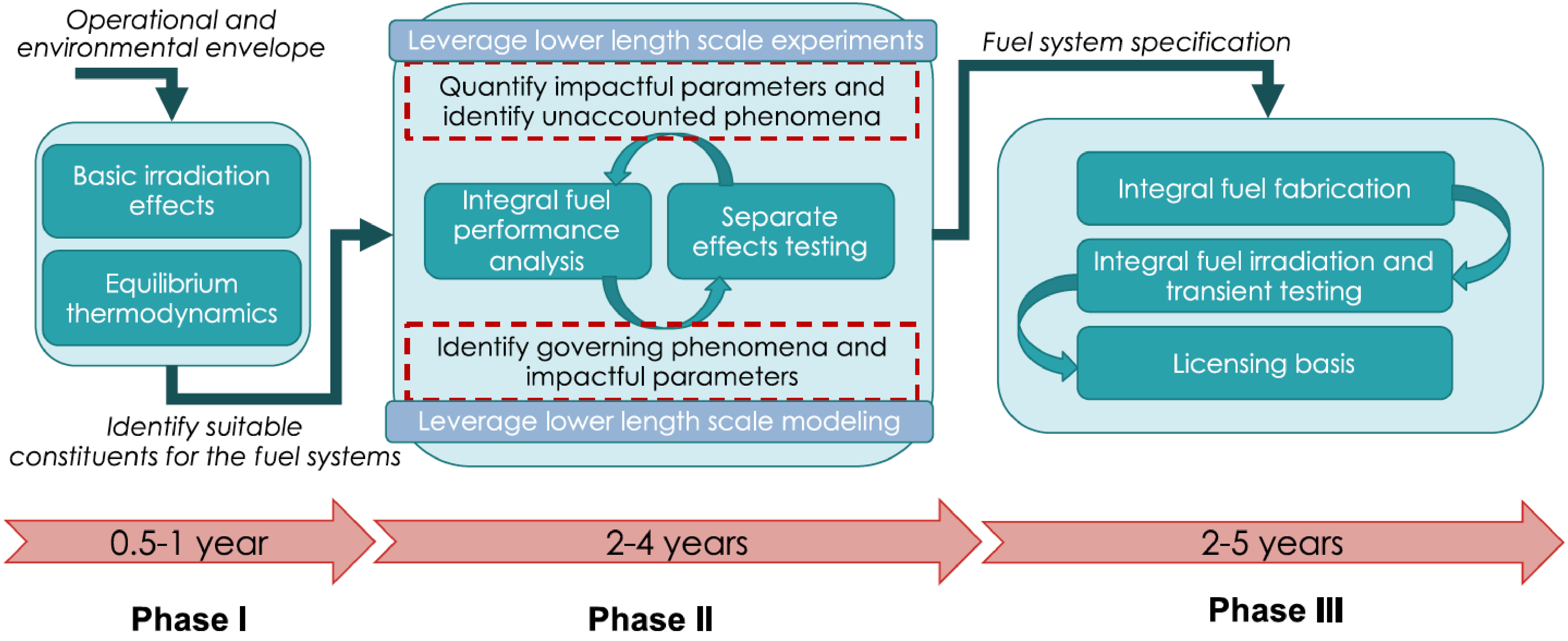


* *Figure from the AFQ white paper*

AFQ Methodology Has Three Phases and Facilitates Judicious Use of Resources

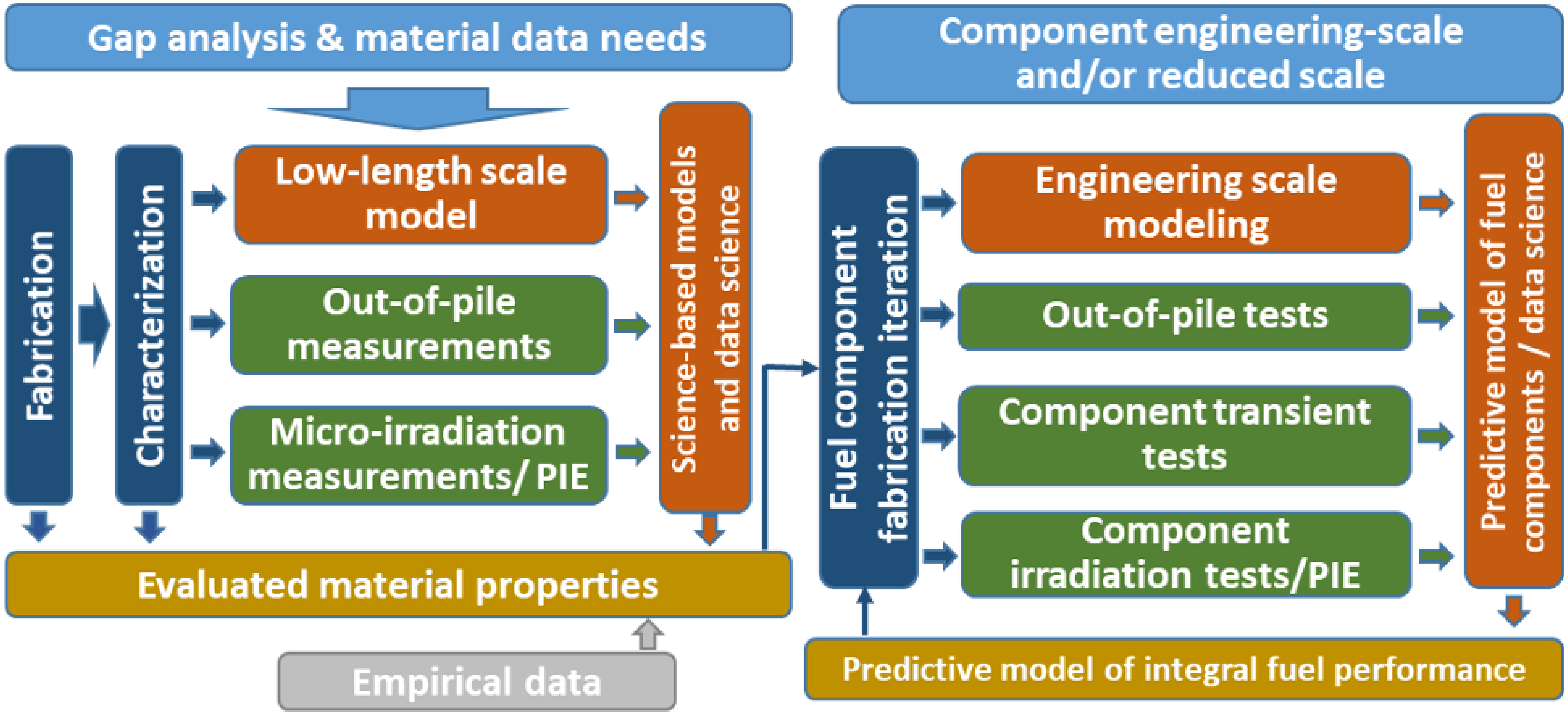
- Phase 1: Design phase
  - Design of fuel system takes advantage of physics-based understanding
  - Assessment identifies gaps and early PIRT analysis
- Phase 2: Detailed analysis and evaluation, involving targeted experiments and dedicated simulations to model constituent behavior
  - Use of M&S in designing experiments and enhancing analysis leads to efficient acquisition of necessary data
- Phase 3: Integral irradiation tests
  - Design of complex integral experiments is better informed by physics-based understanding from separate effects tests
  - Use of AFQ methodology enables quality data and bounding of uncertainties

Employing Modern Tools of AFQ Produces the Most Time Savings During Phase 1 and 2 and Contributes to Higher Quality Data in Phase 3



** Figure from Terrani, et al., JNM 539 (2020) 152267*

Phase 2: Focus on Obtaining Data Through Targeted Experiments and Appropriate Use of Modeling and Simulation



* *Figure from AFQ white paper, and Bolin et al., GA report* 30533R00003 *submitted to DE-NE0008331*

Summary

- **The AFQ methodology** offers a path to qualify new nuclear fuels in a timely and cost-effective way by **leveraging the most advanced M&S and experimental tools that are available today**
- The AFQ methodology is a **suggested guide** to the qualification of new nuclear fuels
- The AFQ methodology **must be tailored for the specific reactor type**, **fuel form, and safety case**
- **Best-practice updates** of AFQ implementation can only be enabled by **shared experience and data from specific user applications**

Path Forward

- Conduct **technical workshops** to share knowledge on the **development, validation and implementation** of **AFQ tools**

- Establish **DOE funding opportunities** to industry, labs and academia for **developing and validating AFQ tools for specific case studies**

- Continue **engagement with the NRC** for **acceptance and adoption of the AFQ methodology** as a valid, optional toolkit for accelerating advanced fuel qualification

- **The AFQ Working Group welcomes the mention of AFQ in the NRC's Draft NUREG-2246, "Fuel Qualification for Advanced Reactors".** This is a great first step in including a high-level summary of the AFQ methodology and stating that "…the AFQ process appear to be consistent with the considerations in the experimental data assessment framework…"

Thank You!

Ron S. Faibish

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Update on the Development of Part 53 Key Regulatory Guidance for Tailored Operator Licensing and Flexible Staffing

Juan Uribe, NRR/DANU
Jesse Seymour, NRR/DRO
Maurin Scheetz, NRR/DRO
November 10, 2021

Agenda

- Background
- Part 53 Tailored Operator Licensing Guidance Development
- Part 53 Flexible Staffing Guidance Development
- Questions

Risk-Informed Thinking

Key messages:

- If advanced reactor designs present very low radiological risk, then current regulatory framework 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**.

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

> 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

Released March 25, 2021 (ML21069A003)

White Paper: **Key Components**

- Rule language
- Supporting Guidance
  - Scalable HFE
  - Tailored Operator Licensing
  - Scalable Staffing

Legend	
	Ongoing Activities
(yellow)	Project under development
(dark blue)	Topics addressed via rule language
(red)	Main document – Operations Roadmap

Innovative Part 53 Rule Language

- Subpart F - Covers the areas of staffing, training, personnel qualifications, and human factors.
- Preliminary Subpart F rule language publicly available via ADAMS at ML21267A006.
- Discussed at 10/26/21 Advanced Reactor stakeholder meeting.
- Introduction with ACRS late September (9/23/2021) and full discussion late November*.

*tentative schedule

Scalable Human Factors Engineering

- Discussed at 09/29/2021 Advanced Reactor Stakeholder meeting.
- NRC/BNL – Purpose is to develop a method for scaling the scope and depth of HFE reviews for advanced reactors.
- 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 plans to issue a white paper with guidance on scalable HFE reviews by December 2021.

NRC Agreement Number: 31310019N002
NRC Task Order Number: 31310020F0028
Technical Letter Report No. F0028-04

Human Factors Engineering Technical Support Services Guidance Development: Process and Guidance Development for Technical Review of Non-Large Light-Water Reactors

Technical Letter Report for Subtask E

Development of HFE Review Guidance for Advanced Reactors

Prepared For:

Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Prepared By:

John O'Hara[1], Stephen Fleger[2], David Desaulniers[2],
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Upton, NY 11798

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Washington, DC 20555-0001

October 12, 2021

Released October 12, 2021 (ML21287A088)

Part 53 Operator Licensing

- Two main categories of operators…
  1. Licensed Operators
     - Includes RO and SRO license levels
     - Initial licensed operator training and examination programs require Commission approval
     - Examinations approved by the Commission; may be administered by facility staff
     - Operator licenses issued by the Commission
     - Requalification training program requires Commission approval and includes periodic requalification examinations
     - Simulators used for reactivity manipulations and licensing exams require Commission approval

Part 53 Operator Licensing

- Two categories of operators… (continued)

  2. Certified Operators (non-licensed)

     - Initial training programs <u>and</u> examination programs require Commission approval
     - Examinations developed and implemented by facility
     - Facility administers certifications and issues certificates
     - Program subject to ongoing inspection by Commission
     - Continuing training program requires Commission approval and includes periodic requalification examinations
     - Simulators used in certified operator programs <u>do not</u> require Commission approval

Operator Licensing Guidance

- Part 53 requires that initial training programs, examination programs, and requalification programs be developed using SAT
  - Focus on process/method
  - Less prescriptive than Part 55
- Guidance is under development on how to review and approve these programs
  - Specialized experience under contract to ensure guidance reflects sound competency assessment testing practices

Operator Licensing Guidance

- Facilities propose programmatic elements
  - What knowledge warrants testing
  - How knowledge will be sampled
  - Examination methods used (written, JPM, etc)
  - What constitutes passing performance
- Comparable to basis of current NUREG-1021, but technology-inclusive and "tailored" to each facility
- NRC reviews and approves facility proposed examination process

Operator Licensing Guidance

- Currently under development
- Multiple opportunities for public engagement
- Staff anticipate that specific opportunities for stakeholder engagement via workshops and comment period on draft guidance will be provided as part of ongoing DOE contract
- Intent is to issue draft guidance to accompany proposed rule language in 2022

Part 53 Staffing Requirements

- Part 53 approach to licensed operator staffing allows facilities to propose staffing plans appropriate for their concept of operation
  - No equivalent to 10CFR50.54 (m) control room staffing levels
  - No requirements for the location of operators
- Applicants need to support staffing plans with relevant HFE-based analyses and assessments to demonstrate that safety functions will be maintained
- Approved staffing plan informs the facility's minimum staffing requirements

Flexible Staffing Reviews

- Scope of the staffing review depends on the *type* of staffing required by Part 53:
  - Plants with licensed operators: need to demonstrate how safety functions are maintained by proposed staffing (HFE analyses/assessments)
  - Plants with certified operators: need to detail how proposed staffing supports certain duties (i.e., admin, monitoring, EP, etc.)
    - Certified operators are not credited for event mitigation; this is central to their relative level of requirements
  - Both sets of requirements are performance-based

Flexible Staffing Guidance

- NUREG-1791 provides a structured review method for evaluating 10 CFR 50.54(m) exemption requests
- Staff is drafting guidance to review staffing plans submitted under Part 53 for plants with licensed operators
  - The guidance under development augments the review guidance in NUREG-1791 to facilitate staffing plan reviews for Part 53 applicants/licensees (no exemption necessary)
- Additional guidance may be developed for review of other staffing plans; these may be incorporated into other ARCAP interim staff guidance documents

Questions?

Acronyms Used

- ARCAP – Advanced Reactor Content of Application Project
- DANU – Division of Advanced Reactors and Non-Power Production and Utilization Facilities
- DOE – Department of Energy
- DRO – Division of Reactor Oversight
- EP – Emergency Preparedness
- HFE – Human Factors Engineering
- ISG – Interim Staff Guidance
- JPM – Job Performance Measure
- NRR – Office of Nuclear Reactor Regulation
- RO – Reactor Operator
- SAT – Systems Approach to Training
- SRO – Senior Reactor Operator

Advanced Reactor Stakeholder Public Meeting

Break

Meeting will resume in 10 minutes

Microsoft Teams Meeting
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Physical Security Q&As for Category II Fuel Cycle Facilities

November 10, 2021

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U.S.NRC
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Overview

- Background
- Physical Protection Requirements for Cat II Fuel Facilities
- Q&A Content
- Key Messages

Background

- Current or recent fuel facility licensing actions have been related to preliminary updates to support higher enrichment
- Extent of the timing and scope of category II quantities (e.g. HALEU) fuel licensing actions is developing with some preliminary pre application interactions taking place
- An important component of those upcoming licensing interactions is the physical security of a Category II quantity of special nuclear material (SNM) of moderate strategic significance

Physical Protection Requirement for Category II Fuel Facilities

- Governed by 10 CFR 73.67(d) which presents requirements for each applicant who possesses, stores, or uses SNM of moderate strategic significance

- Existing guidance for the physical security plan is Regulatory Guide 5.59 “Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance”.

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Q&A Content

1. What are the types of SNM that the NRC regulates?

2. What types of facilities possess Category II quantities of SNM? How are these facilities related to advanced reactor fuels?

3. Can a licensee possess some amount of high-enriched uranium (20% U-235 or greater) and still be considered a Category II facility?

4. What are the existing physical protection requirements and current regulatory approach for Category II quantities of SNM?

5. How will the NRC staff provide consistent and transparent Part 73 reviews?

Q&A Content (cont)

6. What information should an applicant be prepared to discuss to ensure productive discussions with NRC during early interactions (e.g., pre-application meetings)?

7. Has the NRC staff identified lessons learned during the recent SHINE and Centrus license reviews?

8. Is SECY-11-0184, “Security Regulatory Framework for Certifying, Approving, and Licensing Small Modular Reactors," applicable for physical security requirements for fuel cycle facilities with Category II quantities of SNM?

9. Does the Physical Security for Advanced Reactors rulemaking include requirements for Category II SNM at facilities other than power reactors?

10. Will the Part 53 rulemaking include new requirements for Category II quantities of SNM?

Key Messages

- Supplemental security measures for the protection of Category II quantities of SNM may be required to address the current threat environment and the changing understanding of the risks associated with facilities possessing Category II quantities of SNM.
- Staff uses a risk-informed analysis on a case-by-case basis to develop appropriate site-specific supplemental security measures, if needed, that would be implemented through license conditions to ensure the security of Category II quantities of SNM.

Key Messages (cont.)

- Supplemental security requirements could include measures to provide greater security or control over material in use and storage and vital equipment
- To ensure a timely and efficient review, applicants planning to possess Category II quantities of SNM should engage with NRC staff early in the licensing process. The early establishment of an information security program allows for more detailed information to be shared expeditiously.

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US MSR Fuel Salt Qualification Process

NRC – Advanced Reactors Stakeholders Meeting

David Holcomb, George Flanagan, and Mike Poore

November 10th, 2021

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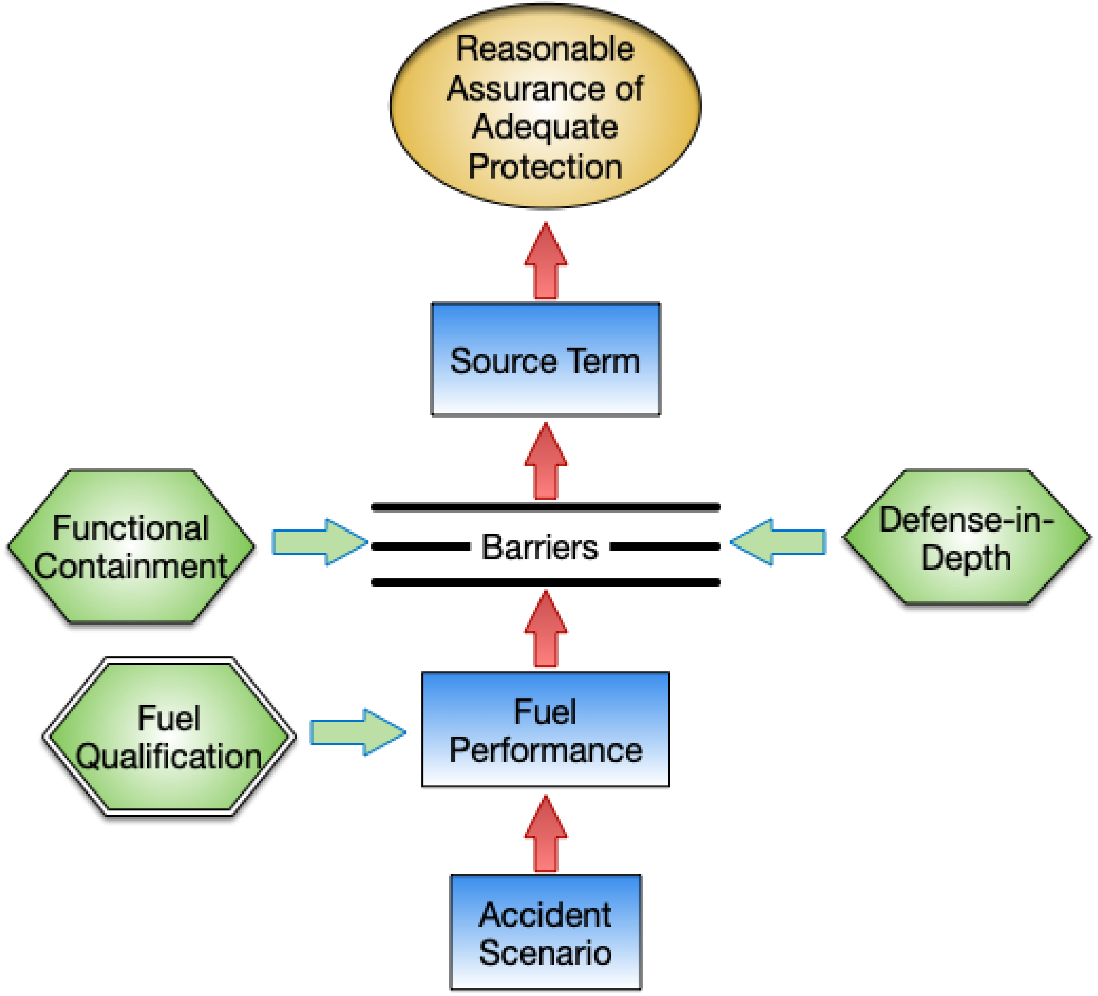
U.S. DEPARTMENT OF ENERGY

Fuel Qualification is an Element in Achieving Sufficient Understanding of Fuel Behavior

*"**Fuel qualification** is a process which provides high confidence that physical and chemical behavior of fuel is sufficiently understood so that it can be adequately modeled for both normal and accident conditions, reflecting the role of the fuel design in the overall safety of the facility. Uncertainties are defined so that calculated fission product releases include the appropriate margins to ensure conservative calculation of radiological dose consequences."*



- NRC Presentation on Possible Regulatory Process Improvements for Advanced Reactor Designs, August 3rd, 2017 (ML17220A315)

Liquid Fuel Has Substantial, Fundamental Differences From Solid Fuel

- Liquid salt fuel
  - Serves as nuclear fuel and primary heat transfer media
  - Must meet requirements for both purposes

Liquid Fuel

- Chemically damageable - may be reparable during use
- Composition may be adjustable during use
- Properties depend on composition and state
- Container breach could release nearly all radionuclides

Solid Fuel

- Mechanically damageable
- Composition set prior to use
- Properties depend on fabrication process

Common Salt Properties and Plant Functions Enable a General Liquid Fuel Salt Evaluation Method

- Specific accident sequences are design dependent
- Basic operational and fundamental safety functions are common to any nuclear power plant
- Halide salt characteristics are common to any MSR
  - High boiling points (low pressure)
  - Low Gibbs free energy (low chemical potential energy)
  - Natural circulation heat transfer properties
- Fuel salt interacts with its container layers via common chemical and physical mechanisms - for example via
  - Thermal energy transfer, chemical reactions, and mechanical processes

Key Issue is "What Constitutes Fuel Salt?"

- Fuel salt does not come in discrete elements (rods or assemblies) and moves independently of its container during normal operations
  - Cladding and fuel assembly structures are qualified as part of solid fuel
- Fuel salt includes all of the material containing fissionable elements or radionuclides that remain in hydraulic communication, but does not include the surrounding systems, structures, or components
  - Salt vapors and aerosols remain part of the fuel salt system until they become adequately trapped
  - Container corrosion products become part of the fuel salt
- Fresh and used fuel salt in on-site storage are within scope

Functional Containment is Important to How MSRs Provide Adequate Radionuclide Retention

- Barrier performance must be degraded to release radionuclides into the environment
  - Performance degradation can occur through failure or bypass
- Fuel salt properties that stress barriers cause them to be more likely to release radionuclides - for example
  - Increased temperature increases radionuclide vapor pressure in cover gas and well as decreasing strength of container
- Different performance requirements for materials normally in contact with salt versus those that only need to withstand accidents

Fuel Salt Boundary Breach Accident Progression Part of Performance Based and Deterministic Fuel Qualification

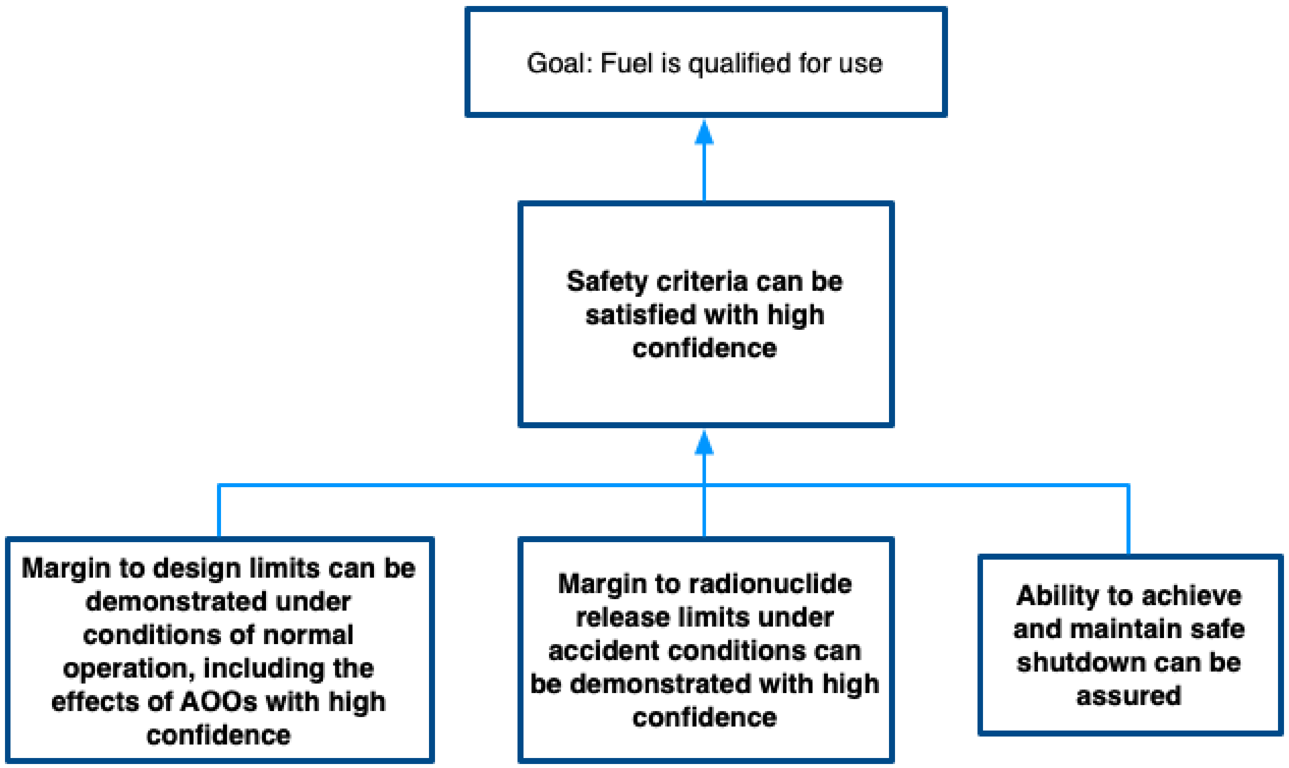
- Multiple locations in the Code of Federal Regulations require evaluation of a postulated fission product release from core into containment
- Fuel salt or cover gas cannot directly stress exterior containment layers without first breaching an inner containment layer
- High radiation and high temperatures immediately outside fuel salt boundary substantially circumscribes characteristics of materials adjacent to fuel salt container
- Focus is on fuel salt properties that must be known to adequately model accident progression and interaction characteristics with materials within containment

Fuel Related Advanced Reactor Requirements Are Similar for Liquid and Solid Fuel

- Example
  - 10 CFR 50.43(e)(1)(i) requires that the performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof
  - Fuel salt thermophysical and thermochemical properties provide the information necessary to model its role in enabling plant safety features to perform safety functions
  - Fuel salt properties vary with both composition and temperature
  - Fuel salt properties need to be determined across the range of temperatures and compositions that span potential operational and accident conditions
  - Quality of the fuel salt property data needs to be sufficient to enable modeling the role of the fuel salt in achieving the plant FSFs

Liquid Salt Fuel Assessment Framework Follows Template Developed for Solid Fueled Advanced Reactors

- Top-down approach used to decompose top level goal of *fuel is qualified* to lower level supporting goals
  - Qualifying fuel develops high confidence that the fuel will adequately perform its role in enabling the facility to achieve its safety objectives
- Lower level supporting goals are further decomposed until clear objective goals are identified that can be satisfied with direct evidence



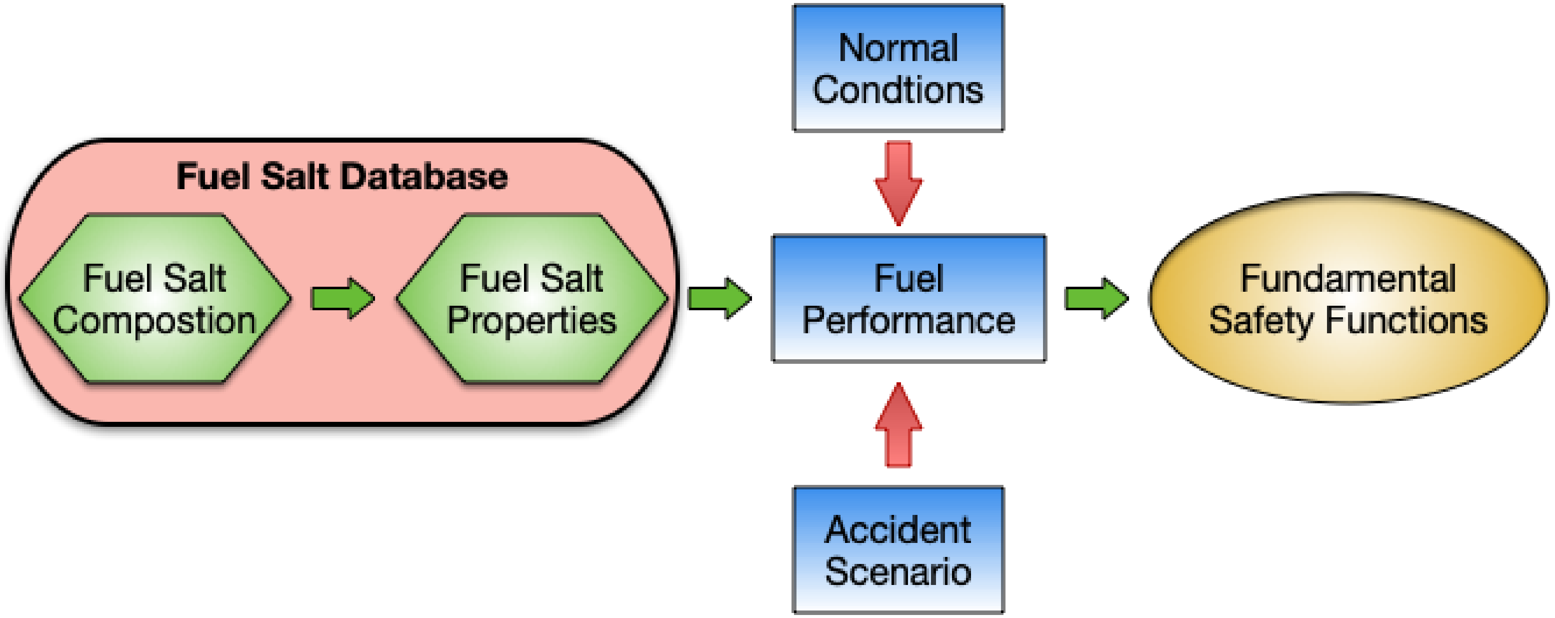
Qualification is Based Upon Understanding the Chemical and Physical Properties of Representative Fuel Samples

- Liquid state significantly changes the physical behavior of fuel
  - Liquids do not accumulate internal stresses
    - No history dependent properties
  - Flow homogenizes fluid properties
    - No position dependent properties
    - No size dependent properties
- Chemical and physical properties are set by elemental composition and temperature
  - Independent of isotopic content

Small minimally-radioactive liquid fuel salt samples provide representative physical and chemical properties

Liquid Fuel Salt Property Database Relates Composition to Physical and Chemical Properties to Aid Developers

- Database development underway sponsored by DOE-NE
  - Database development guided by modeling and simulation
  - Requires appropriate quality assurance for both new and existing data
- Safety evaluations / accident models performed with bounding values to establish acceptable performance range



Fuel Salt Supports the Plant SSCs in Achieving the FSFs and Regulatory Requirements

- Qualification focuses on identification and understanding of fuel salt property degradation mechanisms that occur as a result of irradiation during reactor operation
  - Property repair (composition adjustment) may be incorporated into normal operation
- During normal operations and AOOs fuel salt properties must result in sufficient margin from damage to safety-related SSCs
- Under accident conditions the fuel salt properties must not result in sufficient damage to safety-related SSCs to prevent them from achieving their function

Fuel Qualification Draft NUREG/CR is Available for Review and Comment

- https://www.nrc.gov/docs/ML2124/ML21245A493.pdf
- Suggestions for improvements to the approach can be provided at any time
- Comments and suggestions can be provided to the NRC or ORNL contacts
  - Chris Van Wert, Christopher.VanWert@nrc.gov
  - Richard Rivera, Richard.Rivera@nrc.gov
  - David Holcomb, HolcombDE@ornl.gov

Future Meeting Planning

- The next periodic stakeholder meeting is scheduled for January 19, 2022.
- If you have suggested topics, please reach out to Prosanta.Chowdhury@nrc.gov.