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

Advanced Reactor Stakeholder Public Meeting

April 26, 2023

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 575 470 255#

Advanced Reactor Program

Time	Agenda	Speaker
10:00 am – 10:10 am	Opening Remarks / Advanced Reactor Integrated Schedule	NRC
10:10 am – 10:50 am	Insights from Nuclear Innovation Alliance (NIA) Workshop on Improving Advanced Reactor Licensing Efficiency Advanced Reactor Licensing Review Enhancements	NIA NRC
10:50 am – 11:50 am	Alternative Approaches to Address Population-Related Siting Considerations - White Paper	NRC
11:50 pm – 12:10 pm	NRC Engagement with Tribal Nations	NRC
12:10 pm – 1:25 pm	Lunch Break	All

Time	Agenda (continued)	Speaker
1:25 pm – 1:40 pm	Guidance for Reviewing Facility Training Programs	NRC
1:40 pm – 2:20 pm	Joint NRC/Canadian Nuclear Safety Commission (CNSC) Report on TRI-structural ISOtropic (TRISO) Fuel Qualification	NRC
2:20 pm – 2:35 pm	Break	NRC
2:35 pm – 3:35 pm	CNSC-NRC Memorandum of Cooperation Topic of Safety Classification of Structures Systems and Components: Interim Report	NRC
3:35 pm – 3:40 pm	Future Meeting Planning and Concluding Remarks	NRC

Advanced Reactor Integrated Schedule of Activities

The updated Advanced Reactor Integrated Schedule

is publicly available on NRC Advanced Reactors website at:

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

Advanced Reactor Integrated Schedule of Activities

		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	Guidance		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)
Strategy 6	Communication				

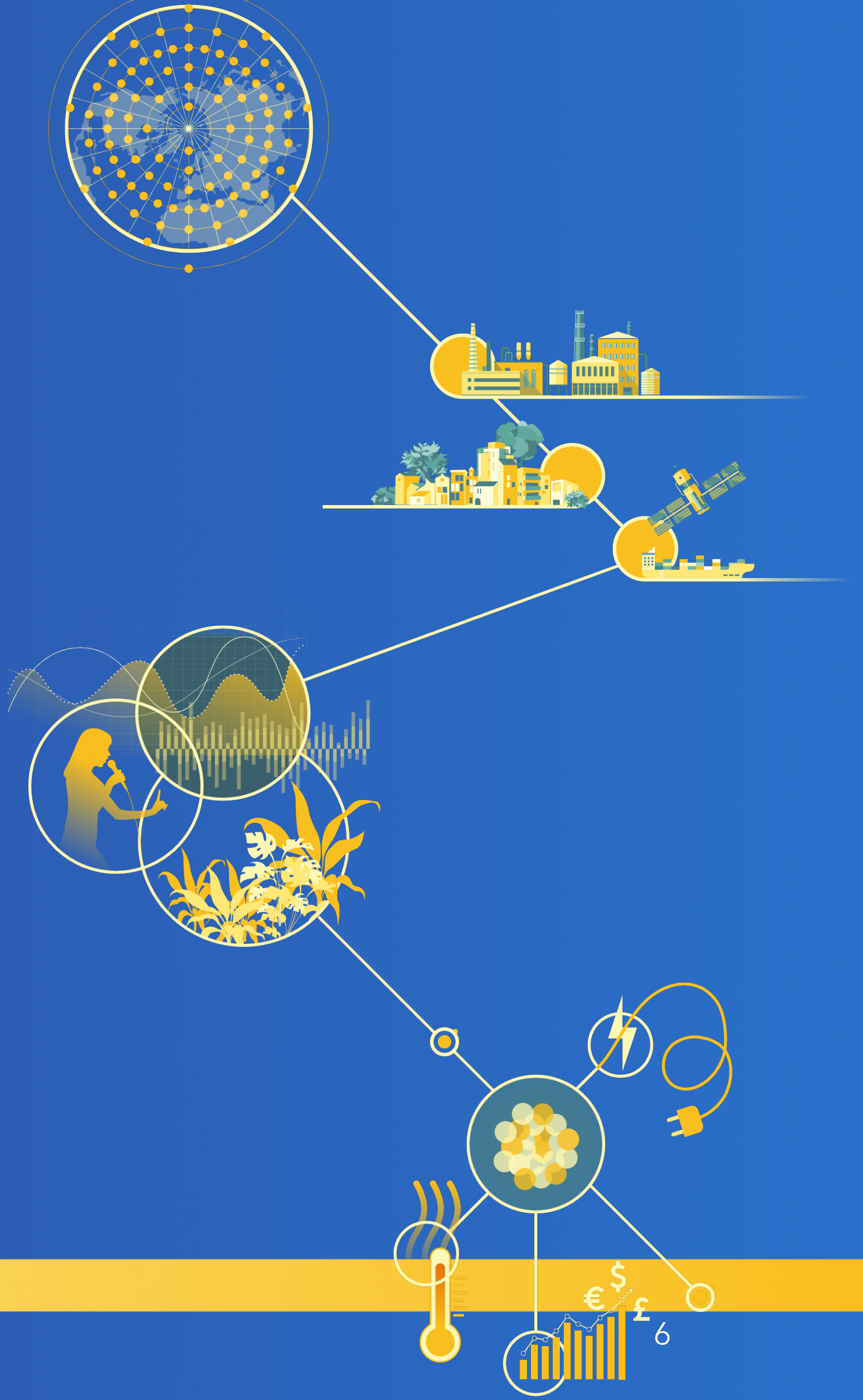
Version 4/20/23

Present Day

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

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

Advanced Reactor Licensing Efficiency Workshop Summary Report



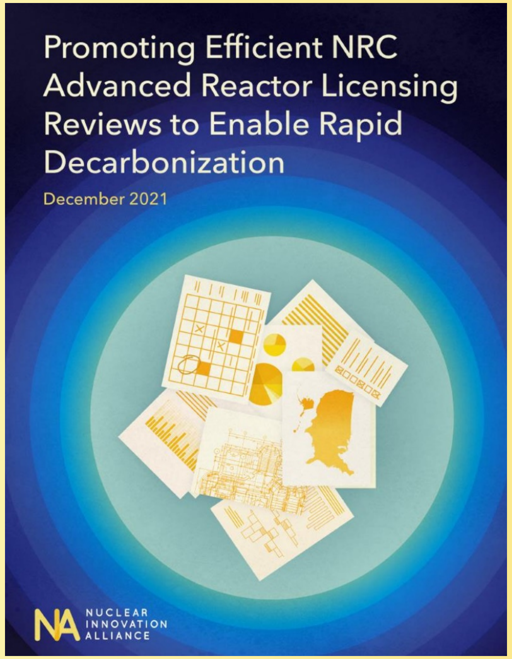
Patrick White (pwhite@nuclearinnovationalliance.org)
NRC Periodic Advanced Reactor Stakeholder Meeting
April 26, 2023

Who is Nuclear Innovation Alliance (NIA)?

- NIA is a "think-and-do" tank working to ensure advanced nuclear energy can be a key part of the climate solution.

- NIA identifies barriers, performs analysis, engages with stakeholders and policy makers, and nurtures entrepreneurship through its Nuclear Innovation Bootcamp.

NIA Licensing Efficiency Workshop was based on prior NIA work with stakeholders on ensuring efficient advanced reactor licensing



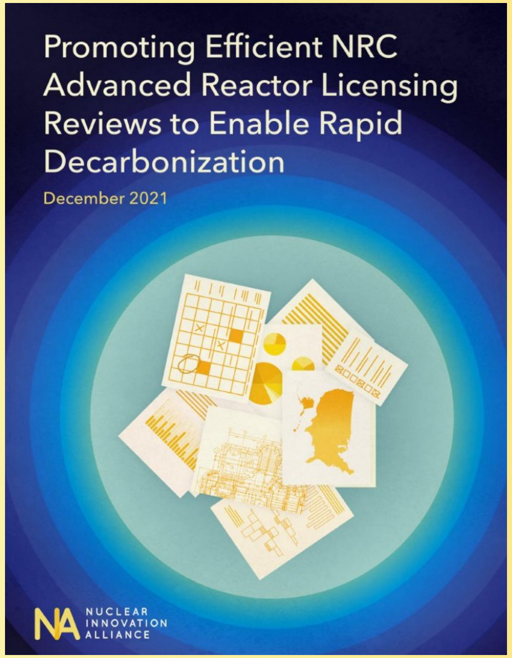
Nuclear Regulatory Commission → **Improving licensing processes**

Advanced Reactor Applicants → **Improving license applications**

Congressional Oversight → **Enabling effective regulation**

Public Participation → **Ensuring regulator accountability**

September 2022 workshop goal was to identify barriers to efficient and effective licensing and share best practices, lessons learned



Identify barriers and solutions to efficient advanced reactor licensing

Nuclear Innovation Alliance Licensing Efficiency Workshop
Workshop Summary Report

NUCLEAR INNOVATION ALLIANCE

Authors:
Patrick White, NIA
NIA Staff

Acknowledgements:
This report is the product of presentations, discussions, feedback, and iteration with advanced reactor developers, power companies, non-governmental organizations, and other stakeholders at the Nuclear Innovation Alliance (NIA)'s September 15, 2022 workshop on improving Advanced Reactor Licensing Efficiency. The Advanced Reactor Licensing Efficiency Workshop was held in accordance with Chatham House Rules which allows sharing of workshop discussions only without direct attribution of specific comments. This report summarizes the major insights from the Advanced Reactor Licensing Efficiency Workshop and is intended to facilitate follow-on discussions with policymakers and stakeholders who were unable to participate in the original workshop. The summary and recommendations presented in this report do not necessarily reflect the views of any specific workshop participant but instead are NIA's synthesis of the workshop presentations and insights.

Please contact Patrick White (pwhite@nuclearinnovationalliance.org) with questions, comments, and feedback.

April 2023

Share best practices and lessons learned (Link to Summary Report)

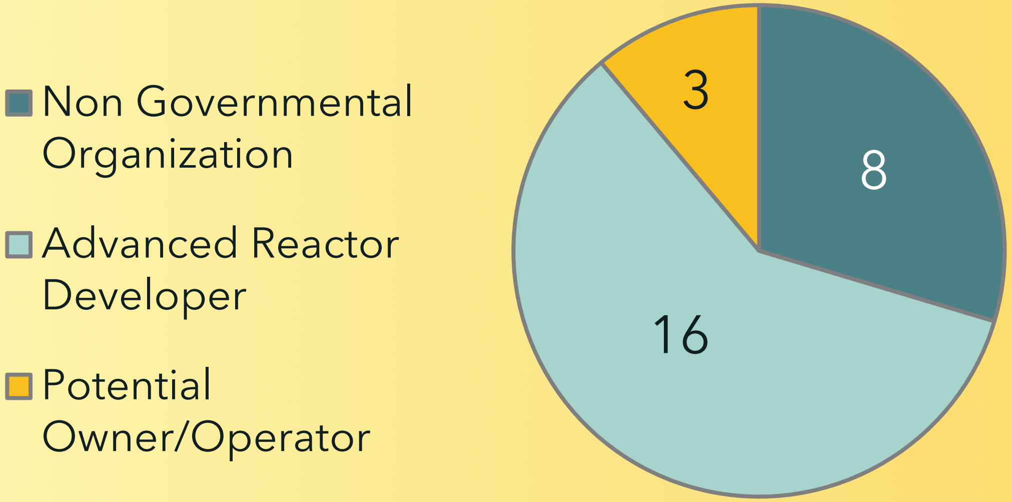
April NRC Periodic Advanced Reactor Stakeholder Meeting

- Share workshop findings
- Discuss recommendations
- Solicit stakeholder feedback
- Discuss possible next steps

Public engagement with NRC on specific recommendations

September 2022 workshop was held under Chatham House Rules to facilitate open, constructive discussion of licensing experiences

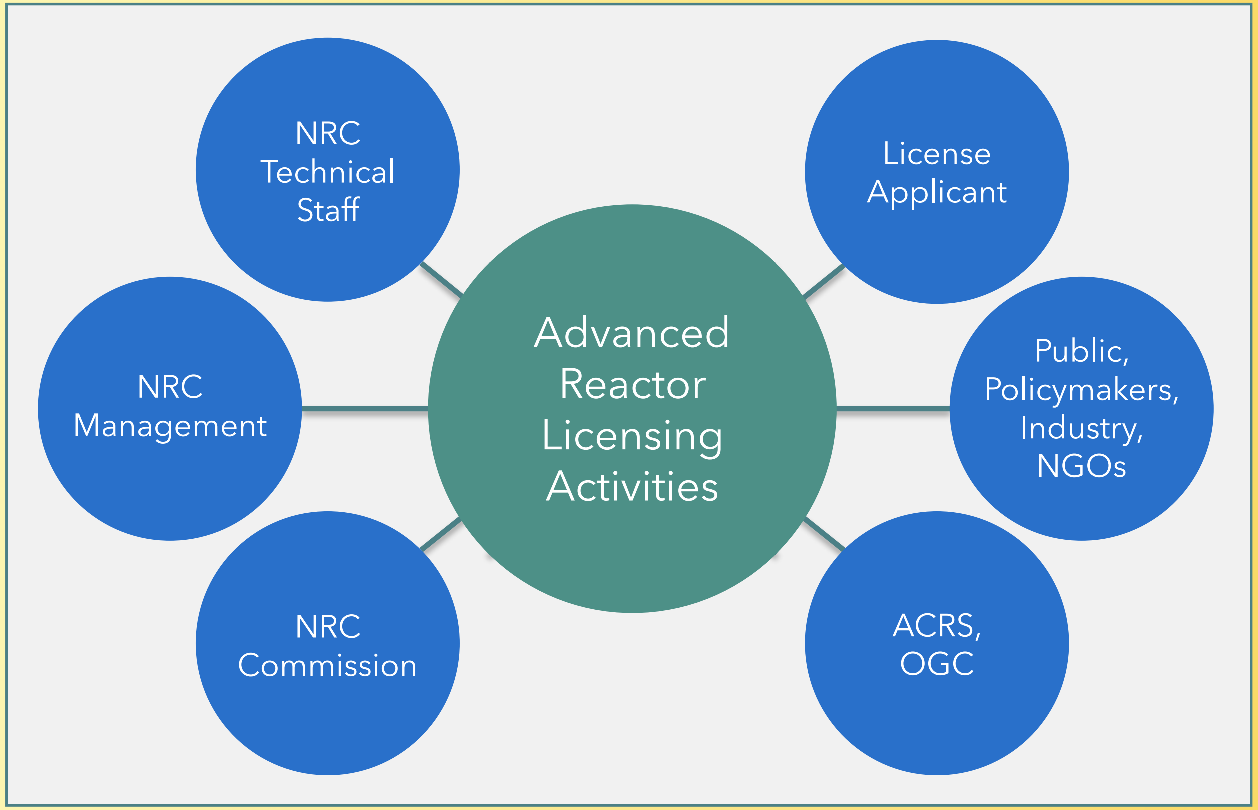
Workshop Participant Affiliation



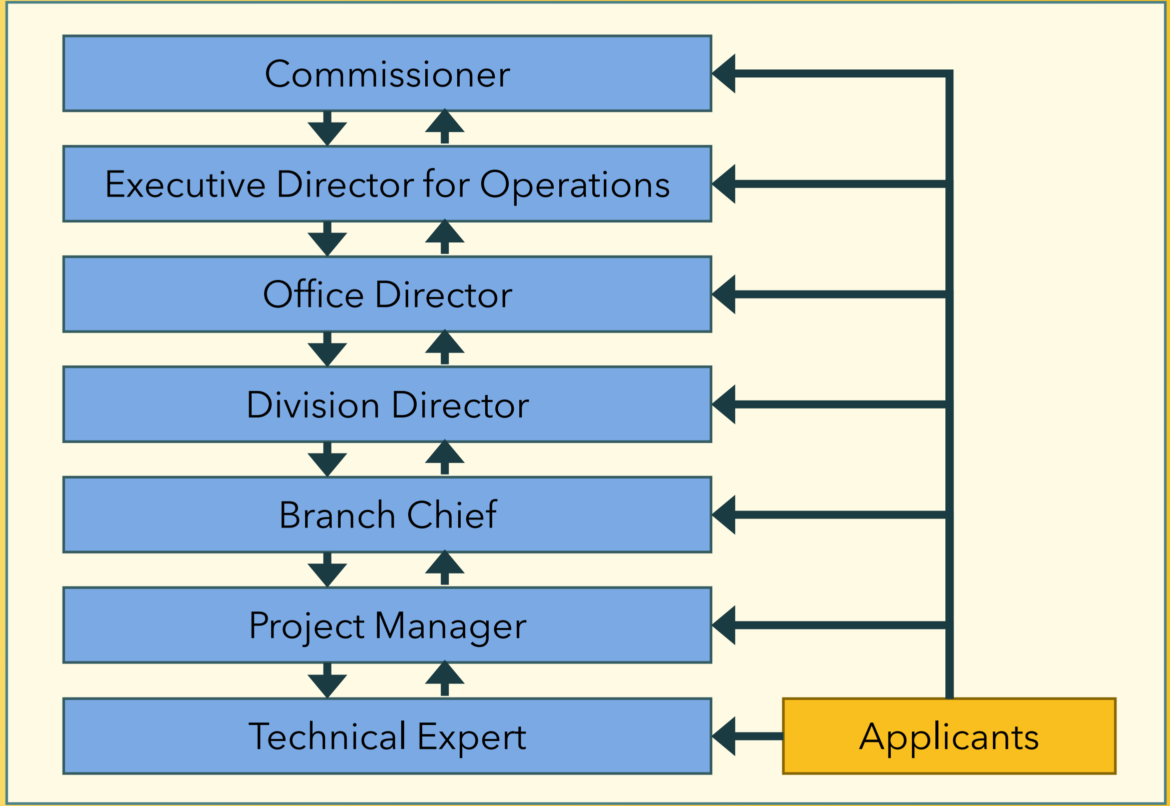
Licensing Efficiency Workshop Sessions
Session 1: Enhancing communication and project management
Session 2: Effectively utilizing regulatory engagement plans and optimizing pre-application interactions
Session 3: Ensuring effective and efficient safety evaluation reviews

Major theme: effective communication is key to efficient licensing

External Communication



Internal Communication



Advanced Reactor Licensing Efficiency Workshop presentations and discussions provided insights across 5 major topic areas

1. Achieving and maintaining alignment between applicant and NRC on the licensing review process and creating clear lines of communication

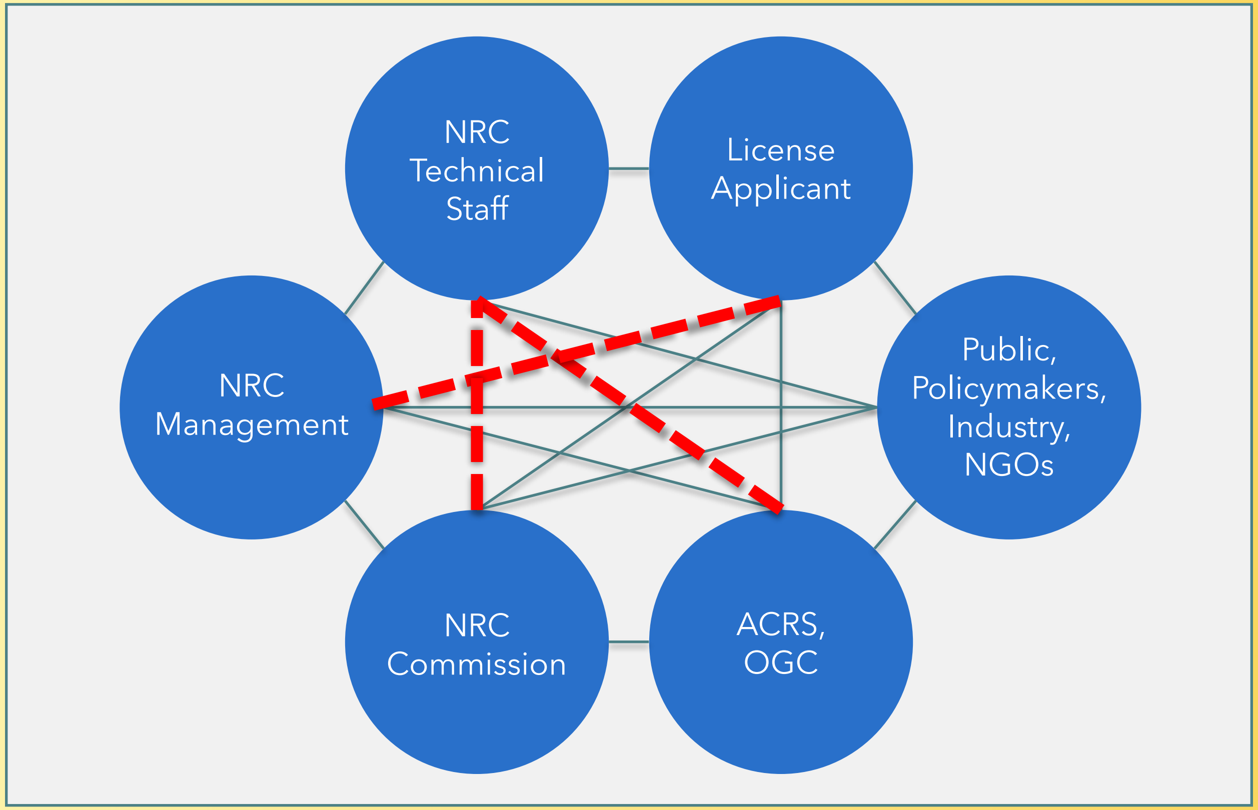
2. Preparing the application content and performing the safety review based on clear, definitive, and consistent expectations

3. Ensuring efficient use of staff resources as the NRC receives an increasing number of advanced reactor license applications

4. Developing processes to identify and resolve challenges encountered during reviews

5. Ensuring uniform understanding and expectations on the role of specific NRC offices and committees in the licensing process

1. Achieving and maintaining alignment between applicant and NRC on the licensing review process and creating clear lines of communication



Communication breakdowns between applicants and NRC or within the NRC can significantly complicate or delay licensing reviews

1. Achieving and maintaining alignment between applicant and NRC on the licensing review process and creating clear lines of communication

Recommendation for Applicants

- Proactively develop lines of communication at all levels as early as practicable
- Maintain lines of communication throughout the review process

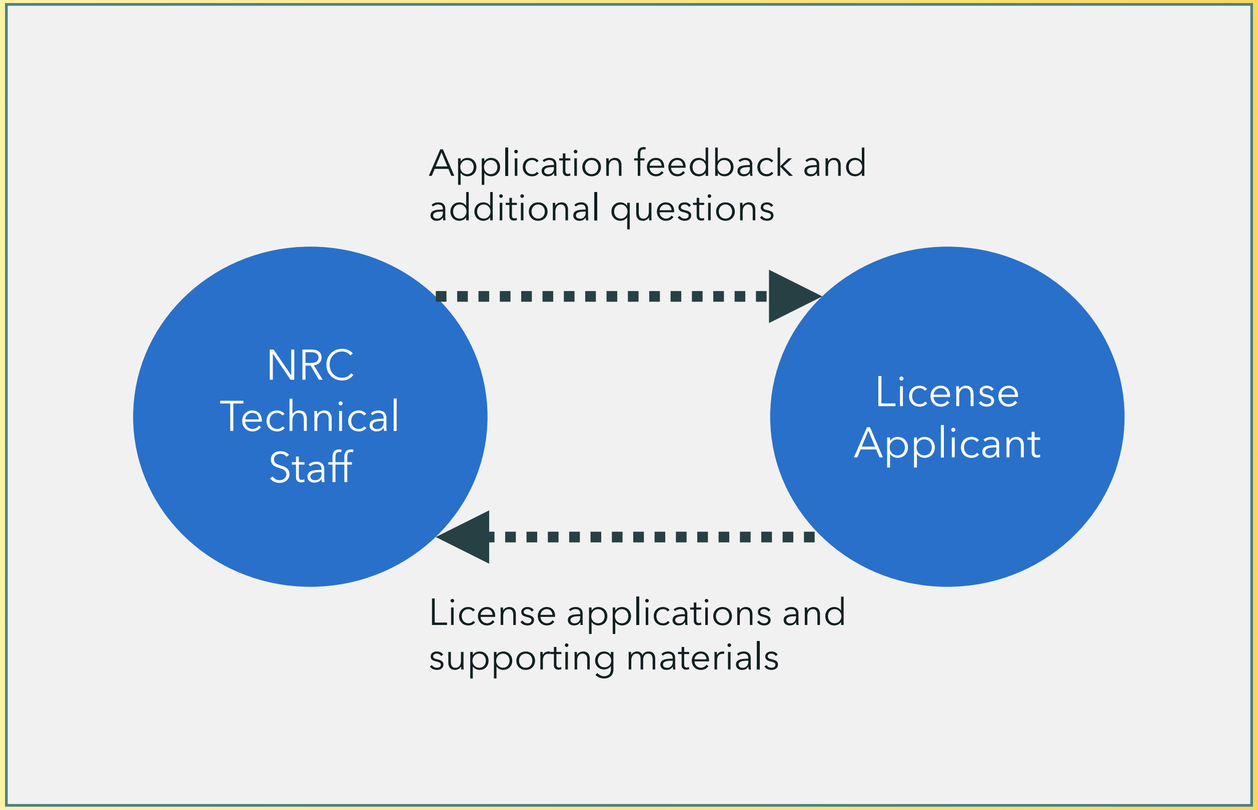
Recommendation for NRC

- Improve internal NRC communication to ensure alignment, clarity, and predictability on technical and policy positions both:
  - Within a specific license review
  - Across different license reviews

Focus: Regulatory engagement plans and specific milestones

- Detailed regulatory engagement plants facilitate staff interaction
- Milestones help hold applicants and NRC accountable on processes
- Communication and plan updates based on licensing progress help maintain alignment

2. Preparing the application content and performing the safety review based on clear, definitive, and consistent expectations



Inadequate or incomplete applications and unclear questions or feedback can result in costly and lengthy iteration cycles between applicants and NRC

2. Preparing the application content and performing the safety review based on clear, definitive, and consistent expectations

Recommendation for Applicants

- Focus on providing information that enables the NRC staff review
- Prepare applications that reduce barriers to the reviewer reaching a safety determination

Recommendation for NRC

- Focus on providing clear feedback and information requests to applicants
- Ensure internal agency alignment on key technical and policy issues

Focus: NRC Licensing Audits

- Licensing audits can facilitate more effective staff reviews of complex issues
- Applicants and NRC should document best practices for licensing audits processes
- Lessons learned should be incorporated into general NRC guidance and process

3. Ensuring efficient use of staff resources as the NRC receives an increasing number of advanced reactor license applications

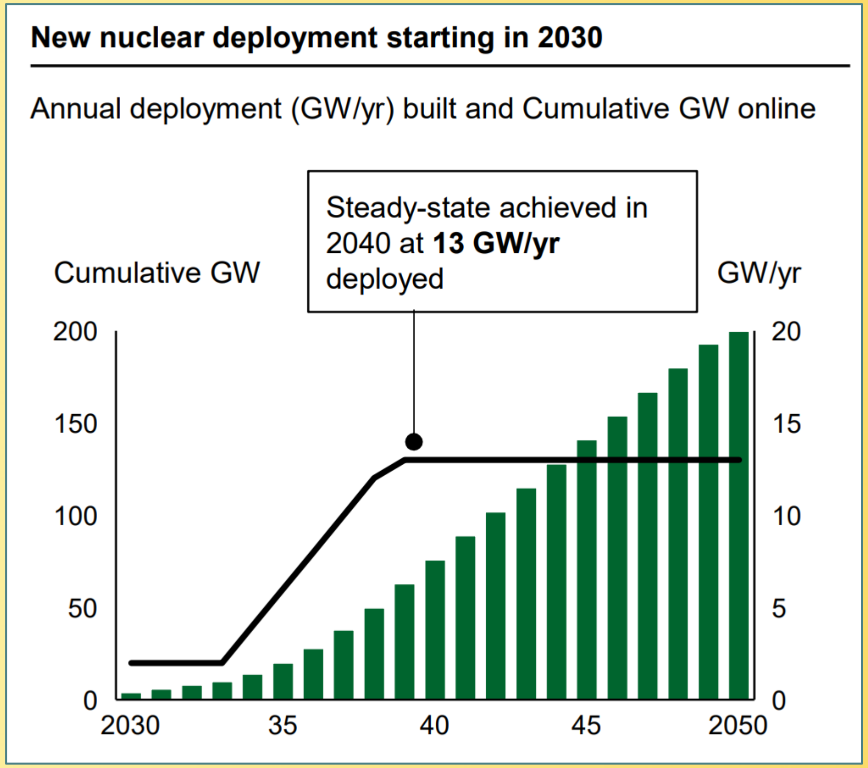


Figure from 2023 DOE Report *Pathways to Commercial Liftoff - Advanced Nuclear*

Participants report some NRC staff resource challenges for current advanced reactor licensing activities, but licensing review workload could increase dramatically to support commercial deployment in the 2030s

3. Ensuring efficient use of staff resources as the NRC receives an increasing number of advanced reactor license applications

Recommendation for Applicants

- Prioritize meeting licensing submittal deadlines provided to NRC staff
- Inform NRC of changing schedule or resource needs for reviews as early as possible
- Facilitate NRC management and planning of resources

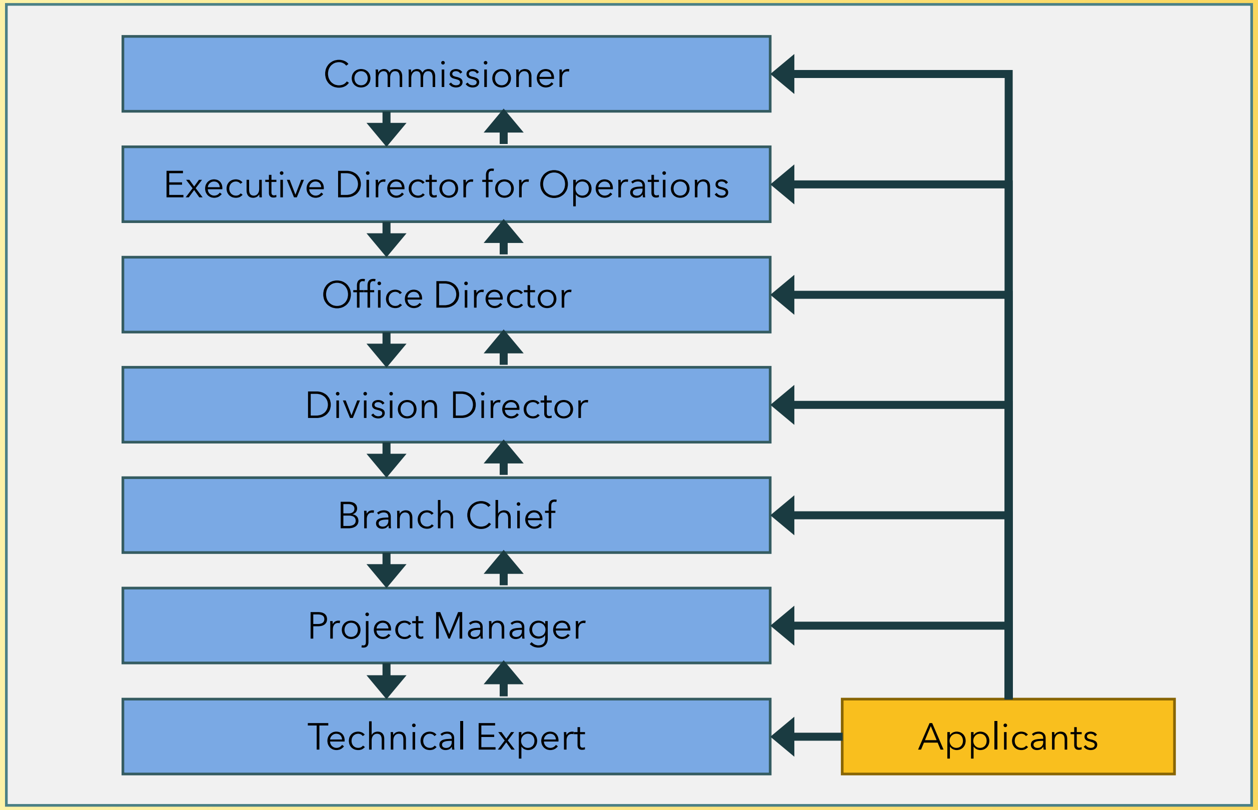
Recommendation for NRC

- NRC management must keep NRC staff accountable for the technical review:
  - Depth,
  - Breadth,
  - Scope, and
  - Regulatory basis

Focus: NRC Project Managers (PM)

- NRC PM performance can have significant effects on licensing process outcomes
- NRC should prioritize the training and organizational management of NRC PMs
- Additional resources, training, and tools could help promote PM excellence

4. Developing processes to identify and resolve challenges encountered during reviews



Applicants and NRC have multiple levels of decisionmakers involved when resolving technical or policy questions, and resolution paths for issues may be unclear

4. Developing processes to identify and resolve challenges encountered during reviews

Recommendation for Applicants

- Proactively share concerns about the licensing process at increasing levels of NRC management
- Avoid the intentional or inadvertent early escalation to senior management or the Commission

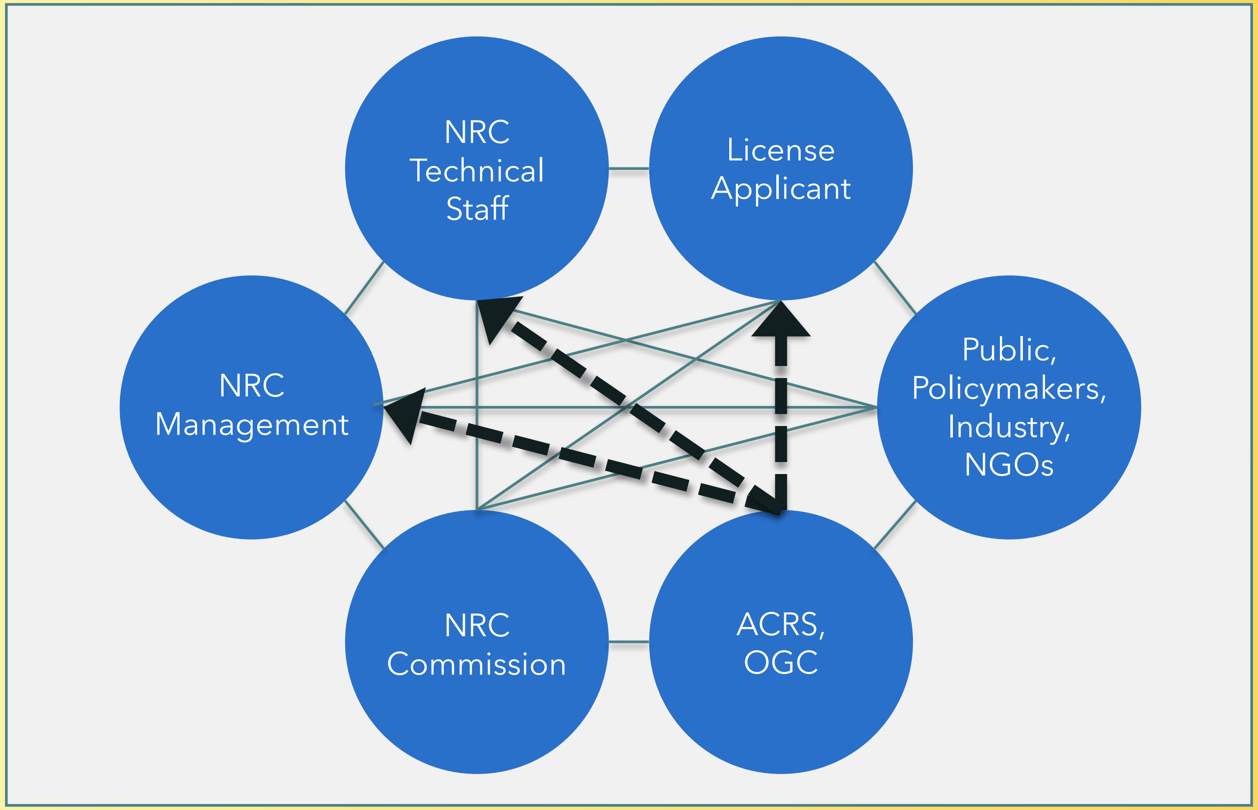
Recommendation for NRC

- Provide regular updates to applicants on both major and minor challenges or questions as they emerge
- Avoid "holding" of concerns or question until the end of a review to discuss with applicants

Focus: Resolving regulation interpretations and issues

- Develop or expand guidance for staff on preliminary decisions
- Assess expedited review procedures for applicants to obtain consistent regulatory interpretations
- Assess an official escalation or appeal process for technical or policy decisions

5. Ensuring uniform understanding and expectations on the role of specific NRC offices and committees in the licensing process



Reviews and decisions from ACRS and OGC can have significant impacts on licensing reviews, but their relationship and interactions with other entities may be unclear to applicants

5. Ensuring uniform understanding and expectations on the role of specific NRC offices and committees in the licensing process

Recommendations for Commission

- Clarify the role of Office of General Counsel (OGC) in licensing reviews so that applicants and staff understand the roles, responsibilities, and scope

- Clarify the role of Advisory Committee on Reactor Safeguards (ACRS) to applicants and staff so they can maximize Committee effectiveness in licensing

Focus: Aligning stakeholder expectations for ACRS reviews

- Clarify expectations for ACRS reviews, interactions with NRC staff and applicants, and the scope of ACRS reviews activities
- Commission should take a more active oversight role on ACRS activities to ensure it maximizes effectiveness

Next steps: soliciting feedback, discussing recommendations, and identifying opportunities for sharing lessons learned, best practices

April NRC Periodic Advanced Reactor Stakeholder Meeting

- Share workshop findings
- Discuss recommendations
- Solicit stakeholder feedback
- Discuss possible next steps

Public engagement with NRC on specific recommendations

Applicant, NRC, and Commission consideration and possible incorporation of report recommendations

Identification of additional opportunities for sharing lessons learned and best practices with applicants, utilities, public, and other stakeholders

Next steps on Licensing Efficiency

Advanced Reactor Licensing Review Enhancements

John Segala
NRR/DANU

Advanced Reactor Stakeholder Meeting
April 26, 2023

NRC Lessons Learned Efforts

The Advanced Reactor Program is informed by stakeholder feedback and several NRC staff lessons learned efforts including:

- New Reactor Licensing Process Lessons Learned Review: 10 CFR Part 52 (ML13059A239)
- Lessons Learned from the NRC Staff's Review of the NuScale Design Certification Application (ML22088A161)
- Response to the NuScale Design Certification Application Lessons Learned Report (ML22294A144)

Enhancing Advanced Reactor Reviews

- Robust Pre-application Engagement
  - Regulatory Review Roadmap (ML17312B567) – Encourages Regulatory Engagement Plans (REPs)
    - NEI 18-06, "Guidelines for Development of a Regulatory Engagement Plan" (non-public NEI document)
  - Pre-application Engagement to Optimize Advanced Reactors Application Reviews white paper
- Expanded Use of Regulatory Audits
  - NRC Office Instruction LIC-111
  - Optimization based on lessons learned
- Optimized use of Requests for Additional Information (RAIs)
  - NRR Office Instruction LIC-115
  - Management review of RAIs before issuance
- Transparency through use of Dashboards

Enhancing Staff Capability and Capacity

- Multidisciplinary core review teams to focus reviews
- Qualification Program for Project Managers
  – Office Instruction updated April 2023
- Building capacity for multiple ongoing reviews
  – Hiring new staff
  – Training staff on advanced reactor technology
  – Use of contractors for flexibility and agility
- Standardized applications will facilitate efficient reviews
- Timely information on industry plans supports effective NRC resource planning

Successfully Implementing Enhancements

- Kairos Hermes Test Reactor Construction Permit (CP) review
  - Successfully executing 21-month review schedule
    - Dashboards
    - Maximizing the use of audits to optimize RAIs
    - Internal project controls
    - Multidisciplinary core review team
- Abilene Christian University Molten Salt Research Reactor CP review
  - Building off the lessons learned from Kairos review
- Pre-application reviews ongoing with multiple developers
  - Regulatory Engagement Plans
  - Successful completion of Topical Report reviews
  - Preapplication assessments enhance readiness and quality of application (NuScale, Atomic Alchemy)

Next Steps

- Continue stakeholder engagement through our periodic advanced reactor public meetings and meetings with developers
- Continue to assess our review processes during ongoing reviews
- Share best practices with prospective applicants
- Continue to make enhancements to internal processes based on lessons learned from ongoing reviews and stakeholder input

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Alternative Approaches to Address Population-Related Siting Considerations

April 2023

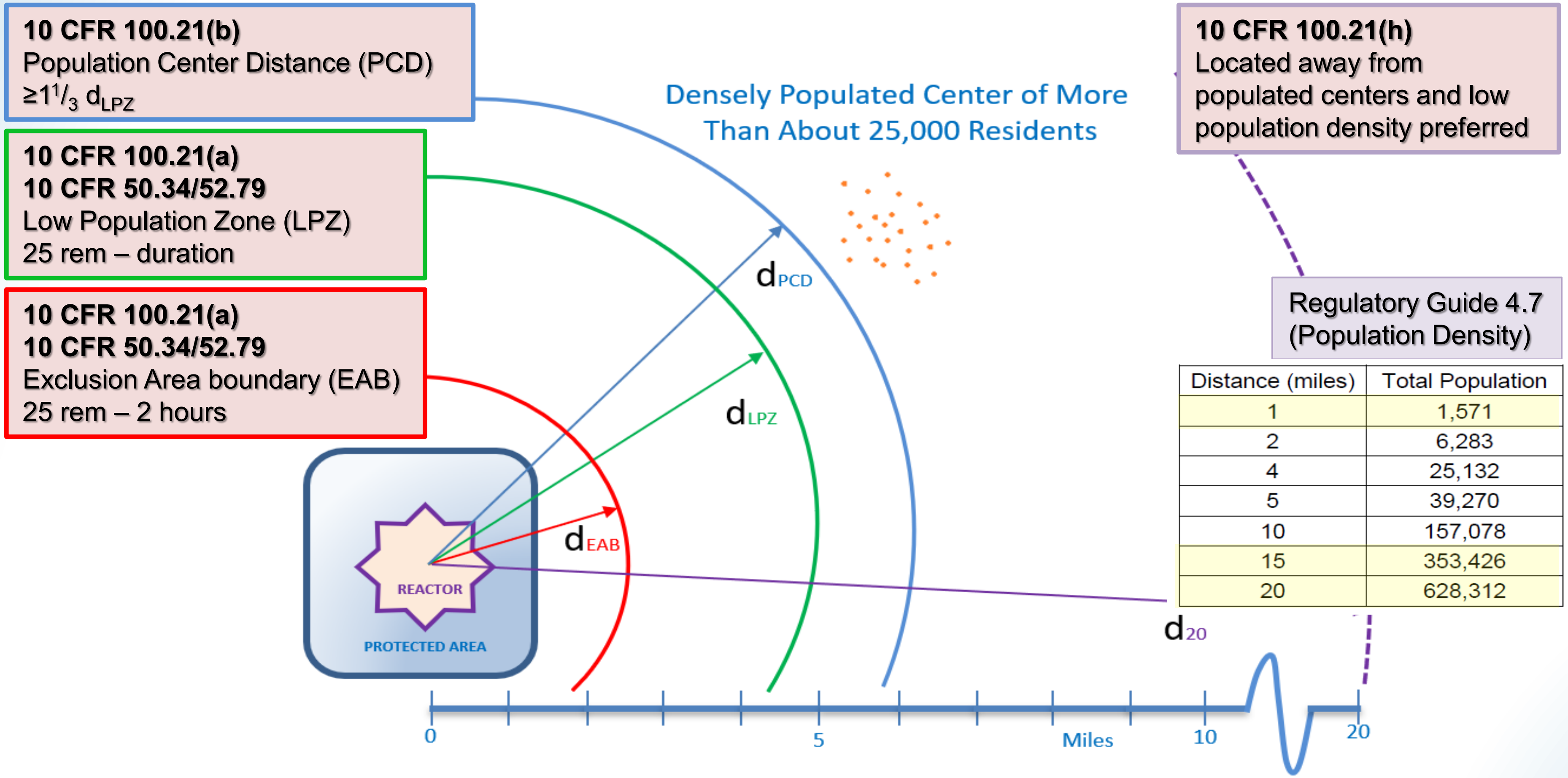
Background

- SECY-20-0045, “Population Related Siting Considerations for Advanced Reactors”

- SRM-SECY-20-0045 dated July 13, 2022
  - ML22194A885

The Commission has approved the staff’s recommended Option 3, to revise the guidance in Regulatory Guide 4.7, “General Site Suitability Criteria for Nuclear Power Stations,” related to Title 10 of the Code of Federal Regulations Part 100, “Reactor Site Criteria,” Section 100.21(h). That provision states that reactor sites should be located away from very densely populated centers and that areas of low population density are generally preferred. The revised guidance will provide technology-inclusive, risk-informed, and performance-based criteria to assess population-related issues in siting advanced reactors. With respect to the traditional dose assessment approach, the staff should provide appropriate guidance on assessing defense-in-depth adequacy and establishing hypothetical major accidents to evaluate.

Background – Requirements/Guidance



Distance (miles)	Total Population
1	1,571
2	6,283
4	25,132
5	39,270
10	157,078
15	353,426
20	628,312

d_{EAB} – radial distance to the exclusion area boundary (EAB)

d_{LPZ} – radial distance to the outer boundary of the low population zone (LPZ)

d_{PCD} – population center distance (PCD) to the nearest boundary of a densely populated center

d_{20} – 20 mile outward radial distance (population density not to exceed 500 persons per square mile – RG 4.7)

Background

- Two potential issues identified:
  1) 500 persons per square mile (ppsm) out to a distance of 20 miles
  2) 500 ppsm close to reactor site used for small communities

  – Background and references in ORNL/TM-2019/1197 (ADAMS Accession No. ML19192A102)

- Staff developed several options for consideration:
  – Option 1 – Status Quo
  – Option 2 – Source Term Factor
  – Option 3 – Offsite Dose Calculation
  – Option 4 – Develop Societal Risk Measures

Option 3 (Offsite Dose Calculation)

Description

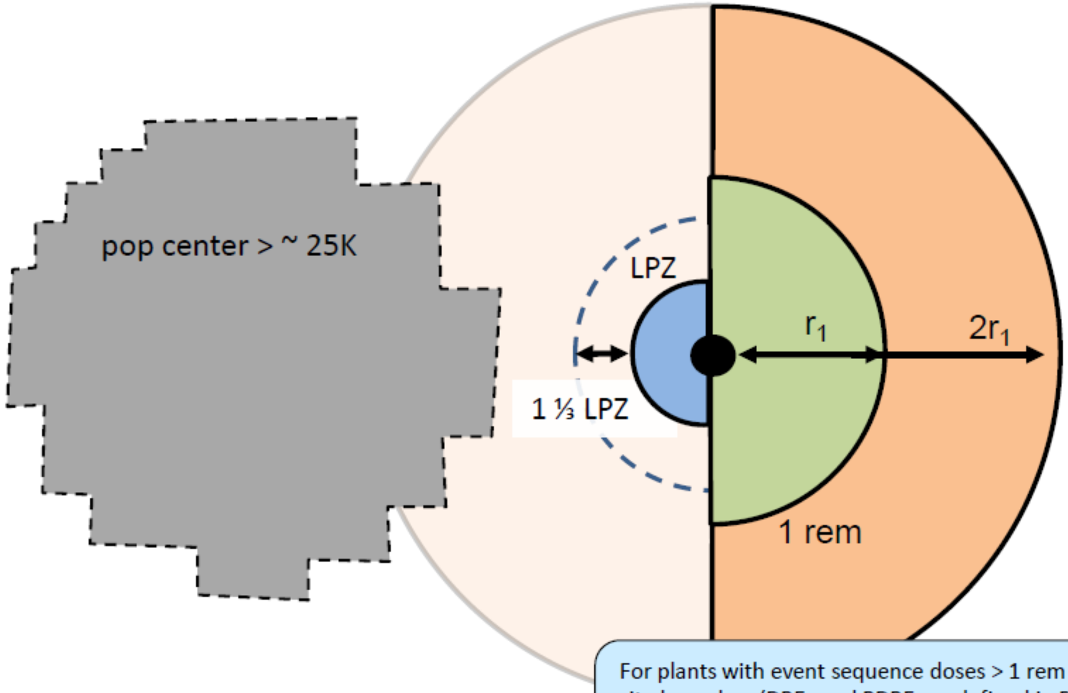
- Maintain EAB and LPZ for event sequence doses of 25 rem TEDE over 2 hours and course of event respectively
- Maintain distance from densely populated center of more than about 25,000 residents
- For plants with event sequence doses > 1 rem TEDE over a month beyond the site boundary (DBEs and BDBEs as defined under licensing modernization project (LMP)), population density < 500 ppsm over the radial distance equal to twice the radius at which 1 rem over a month is estimated

Option 3 – Example Cases

Case 1:

Event Sequences with Offsite Doses > 25 rem over course of event

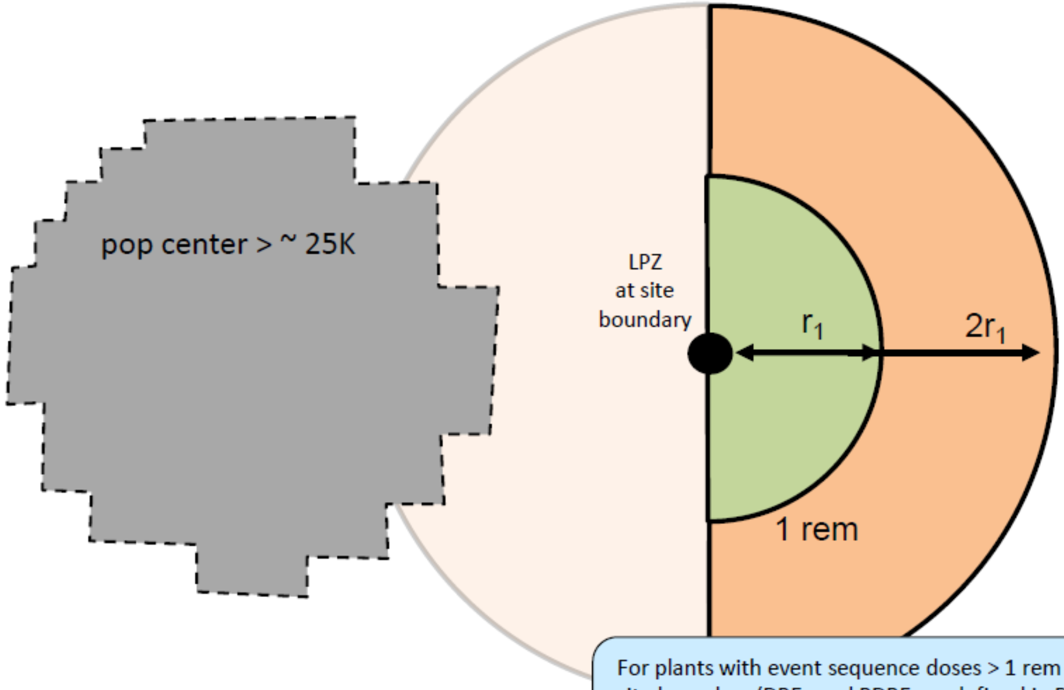
Event Sequences with Offsite Doses > 1 rem over the month following event



For plants with event sequence doses > 1 rem over a month beyond the site boundary (DBEs and BDBEs as defined in DG-1353), population density < 500 ppsm over the radial distance equal to twice the radius at which 1 rem over a month is estimated

Option 3 – Example Cases

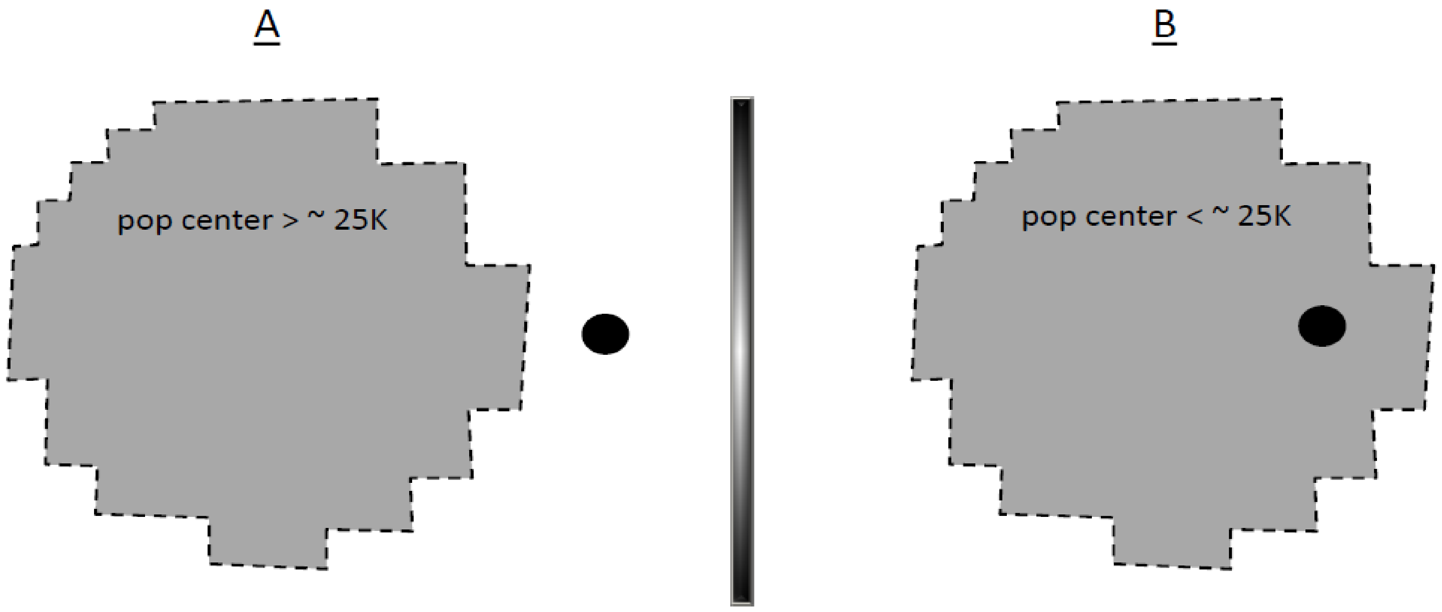
Case 2:
<u>No</u> Event Sequences with Offsite Doses > 25 rem over course of event
Event Sequences with Offsite Doses > 1 rem over the month following event



For plants with event sequence doses > 1 rem over a month beyond the site boundary (DBEs and BDBEs as defined in DG-1353), population density < 500 ppsm over the radial distance equal to twice the radius at which 1 rem over a month is estimated

Option 3 – Example Cases

Case 3:
No Event Sequences with Offsite Doses > 25 rem over course of event (LPZ at site boundary)
No Event Sequences with Offsite Doses > 1 rem over the month following event



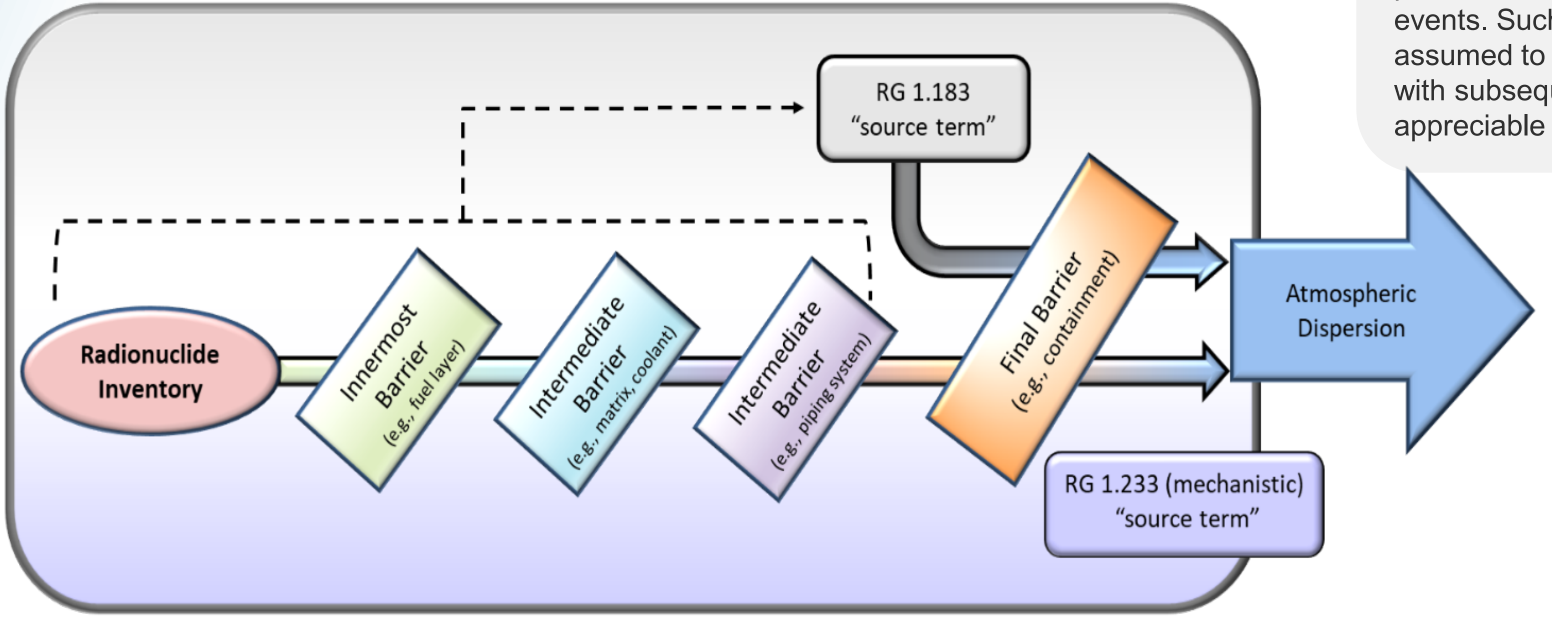
Population center distance means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents

White Paper

- Prepared to support public meetings and discussion of future changes to Regulatory Guide 4.7
- Preliminary approaches for
  - Non-light water reactors under LMP-type methodology
  - Light water reactors under traditional methodology
  - Non-light water reactors under traditional (non-LMP) methodology
- Distinctions between:
  - Analyses related to estimated doses at EAB/LPZ
  - Analyses related to alternative to existing population density guidance (500 ppsm out to 20 miles)

"Source term" for "siting analysis"

Footnote (6) – 10 CFR 50.34
The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.



- Additional Information provided in:
  - Regulatory Guide 1.183
  - Regulatory Guide 1.233

Non-LWR under LMP

- LMP approach for non-LWRs was primary focus of SECY-20-0045
- Preliminary white paper methodology
  - Analyses related to estimated doses at EAB/LPZ
    - Design Basis Accidents
  - Analyses related to alternative to existing population density guidance (500 ppsm out to 20 miles)
    - Design Basis Events and Beyond Design Basis Events
      - Outputs used to determine distance at which an event results in 1 rem TEDE over 30 days

LWR under traditional approach

- SECY-20-0045 mentions using traditional approach (RG 1.183)
- SRM directed staff to provide guidance on assessing defense-in-depth adequacy and establishing hypothetical major accidents to evaluate
- Preliminary white paper methodology
  - Analyses related to estimated doses at EAB/LPZ
    - Regulatory Guide 1.183
  - Analyses related to alternative to existing population density guidance (500 ppsm out to 20 miles)
    - Regulatory Guide 1.183
    - Accounting for potential containment performance under severe accident conditions

Non-LWR under traditional approach

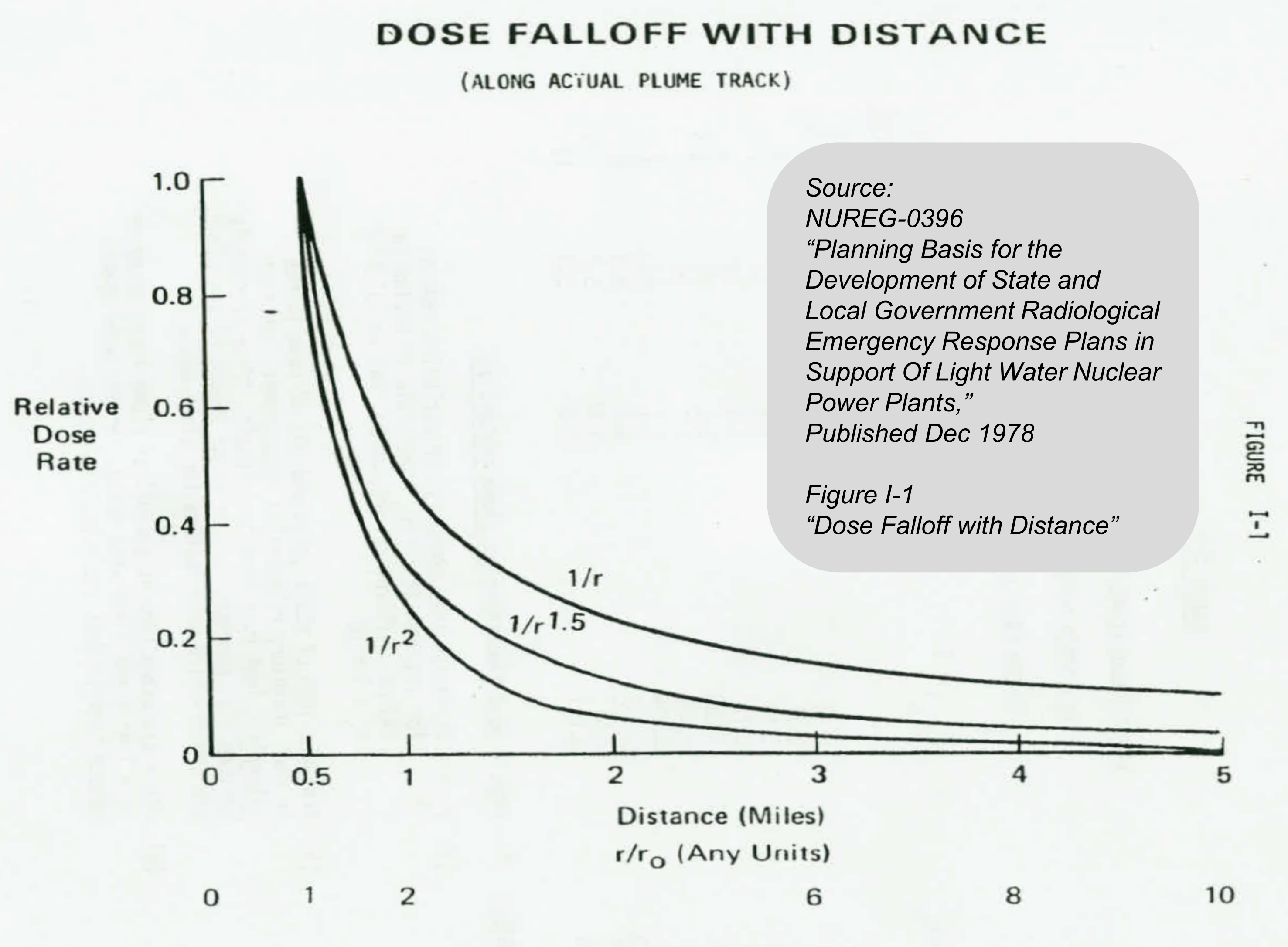
- SECY-20-0045 mentions using traditional approach (RG 1.183)
- SRM directed staff to provide guidance on assessing defense-in-depth adequacy and establishing hypothetical major accidents to evaluate
- Guidance prepared for non-LWRs relying on containment type design feature as a primary means to limit the release of radionuclides
- Preliminary white paper methodology
  - Analyses related to estimated doses at EAB/LPZ
    - Regulatory Guide 1.183 like analysis for source term used for assessing containment and site-specific information
  - Analyses related to alternative to existing population density guidance (500 ppsm out to 20 miles)
    - RG 1.183 like source term
    - Accounting for potential severe accidents that challenge the containment

Moving forward

- Prepare draft revision 4 to RG 4.7
  - DG 4031
- Publish draft guidance for public comment
  - Target: Fall 2023
- Resolve public comment
- Issue Final RG (revision 4 to RG 4.7)
  - Target: 1st quarter CY 2024

Questions and Discussion

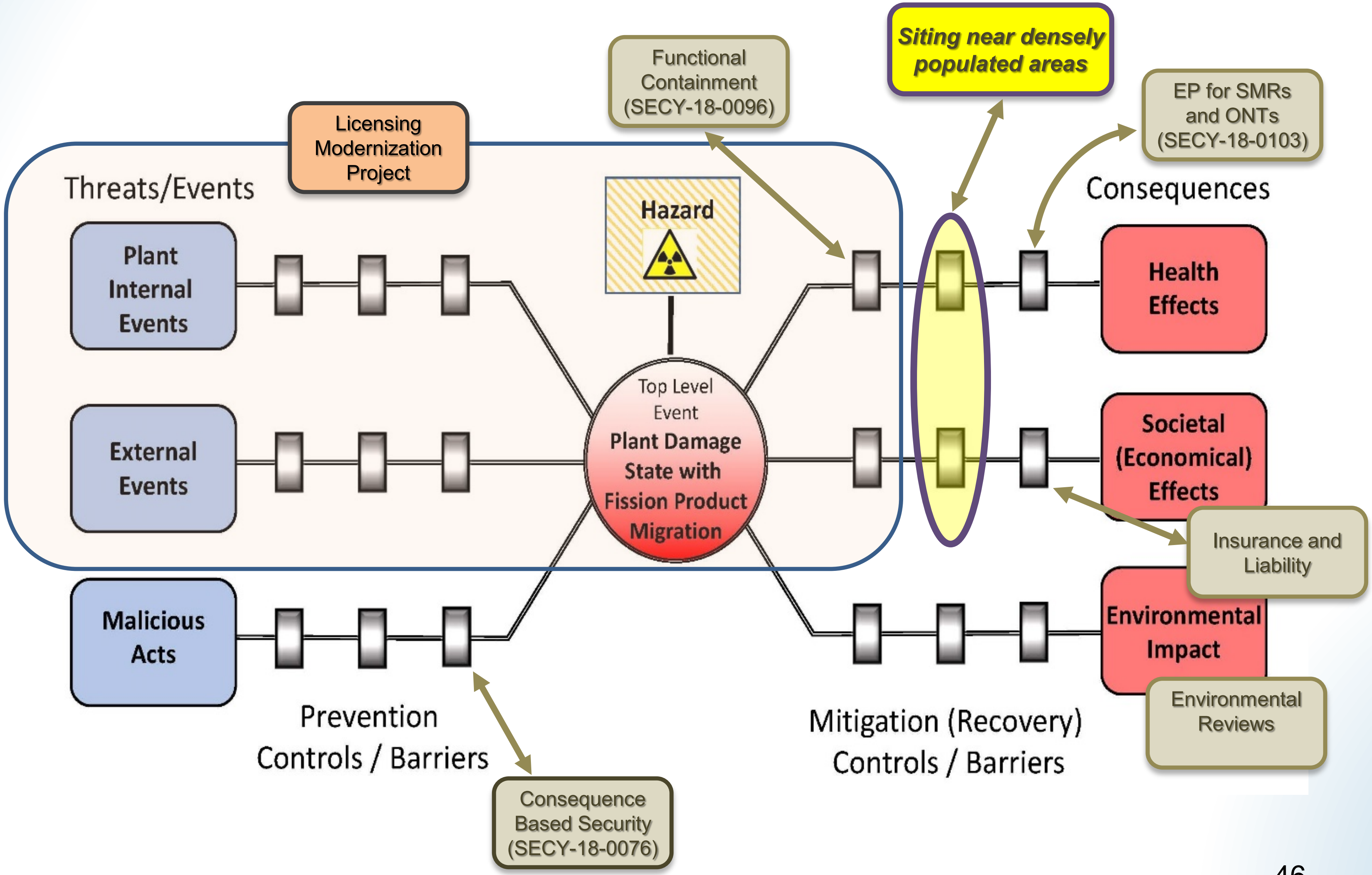
Backup Slide – Dose Falloff



Source:
NUREG-0396
"Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support Of Light Water Nuclear Power Plants,"
Published Dec 1978

Figure I-1
"Dose Falloff with Distance"

Backup Slide – Integrated Approach



Backup Slide – Integrated Approach

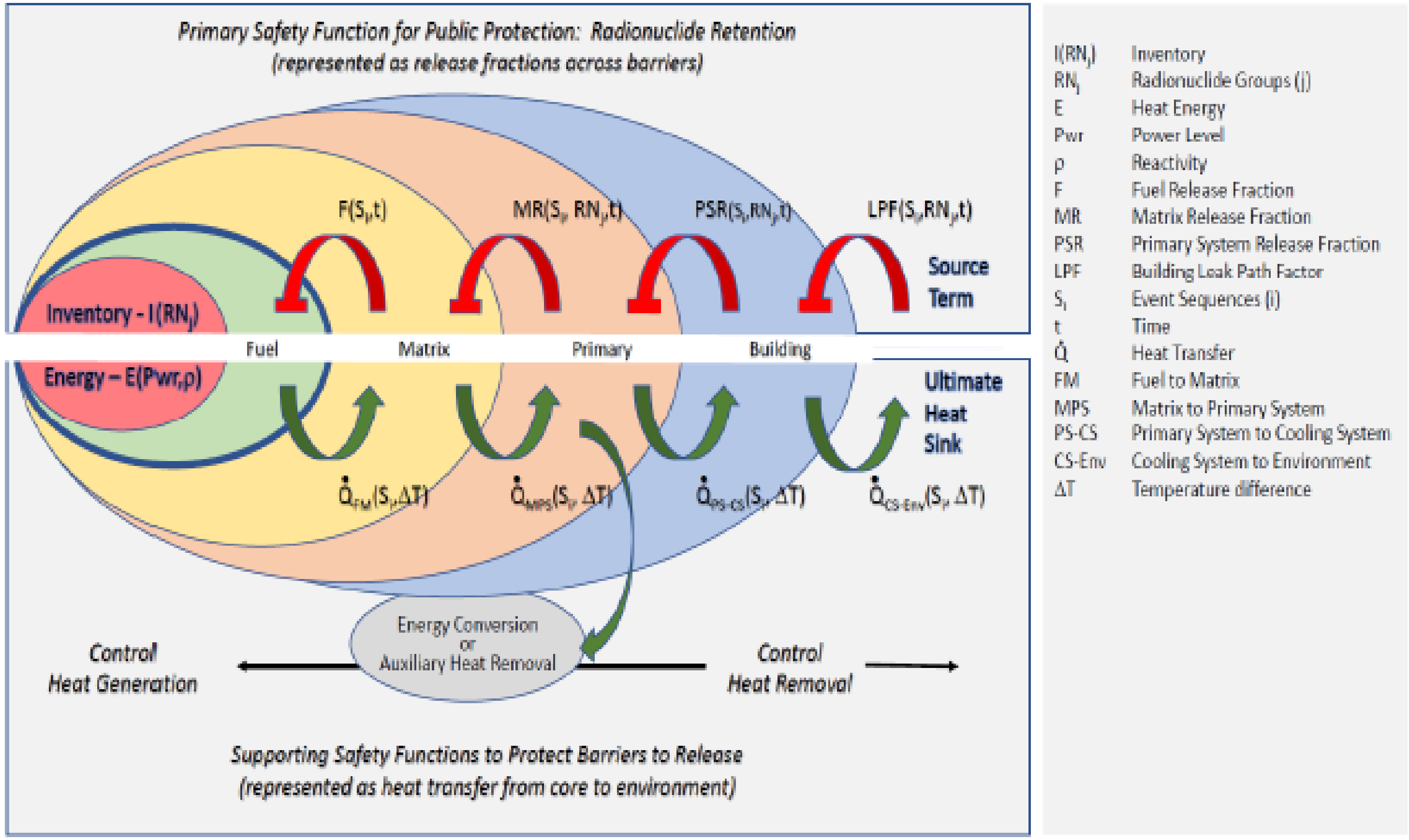
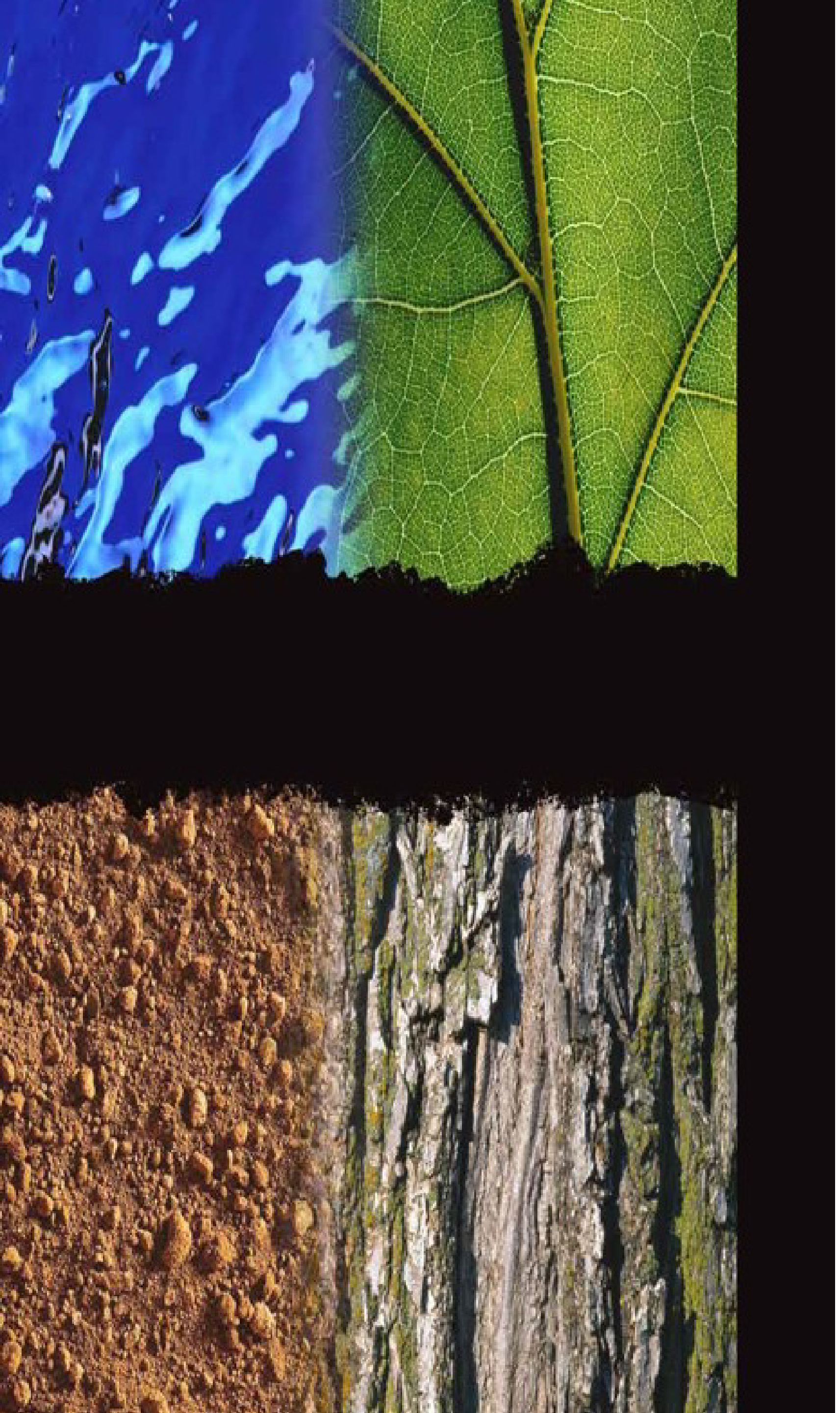


Figure 1 Fundamental safety functions and mechanistic source term

SECY-19-0117: 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



NRC Engagement with Tribal Nations

Kevin Williams, Director
Division of Materials Safety, Security, State, and Tribal Programs

April 26, 2023
Advanced Reactor Public Stakeholder Meeting

Tribal Policy Statement

In 2017, the NRC published the Tribal Policy Statement is centered on the following principles:

1. The NRC recognizes the Federal trust relationship with and will uphold its trust responsibility to Indian Tribes.
2. The NRC recognizes and is committed to a government-to-government relationship with Indian Tribes.
3. The NRC will conduct outreach to Indian Tribes.
4. The NRC will engage in timely consultation, as applicable.
5. The NRC will coordinate with other Federal agencies, as applicable.
6. The NRC will encourage participation by State-recognized Tribes.

This principles guide the NRC's government to government interactions with the Tribal Nations.

The NRC does its part in implementing this duty in the context of our jurisdiction and in honoring treaties.

Licensing Reviews

Consultation

NEPA	SECTION 106
Notice of Intent	
NEPA Scoping	Identify Section 106 Consulting Parties Identify Area of Potential Effect
Prepare Draft Environmental Impact Statement	Identify historic resources within the Area of Potential Effect Identify potential adverse effects Develop measures to avoid, minimize, or mitigate adverse effects
Public Review of Draft Environmental Impact Statement	Publish Draft Section 106 Memorandum of Agreement (as needed) Accept public comments on the Draft Section 106 Memorandum of Agreement (as needed)
Prepare Final Environmental Impact Statement	Respond to public comments and revise Section 106 Memorandum of Agreement (as needed) Execute Section 106 Memorandum of Agreement (as needed)
Record of Decision	

Key Differences in Tribal Consultation between the National Historic Preservation Act Section 106 and the NRC’s Tribal Policy Statement

- [NRC’s Tribal Consultation Information Tool (ML23019A328)]

NRC Tribal Program Contacts

- Kevin Williams, Director
- Email Kevin.Williams@nrc.gov
- Phone: 301-415-3340

- Booma Venkataraman, Branch Chief
- Email: Booma.Venkataraman@nrc.gov
- Phone: 301-415-2934

- Contact the NRC's Tribal Program Team
- Email: Tribal_Outreach.Resource@nrc.gov

- NRC General Contact Information
- https://www.nrc.gov/about-nrc/contactus.html

Questions?

Lunch Break

Meeting will resume at 1:25 pm EST

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 575 470 255#

Facility Training Program Guidance DRO-ISG-2023-04

- This ISG is intended to support both applications under the proposed Part 53 as well as near-term applications under Parts 50 and 52.
- The guidance supports the NRC staff review of the portion of an application associated with the training program for plant personnel, including licensed operator initial and requalification training programs.
- This guidance also facilitates the review of non-accredited training programs at commercial nuclear plants. This guidance may also be used to support training program inspection needs as currently specified in NUREG-1220.
- This guidance covers:
  - Scope of facility training programs
  - The 5 phases of the systems approach to training

CNSC/NRC
TRISO Fuel Qualification Assessment

Draft Final Report

U.S. NRC Advanced Reactor Stakeholders Meeting

April 26, 2023

Kelly Conlon, Canadian Nuclear Safety Commission (CNSC)

Jeff Schmidt, U.S. Nuclear Regulatory Commission (NRC)

Objective and Status

- The Generic Tristructural Isotropic (TRISO) qualification assessment advances the NRC/CNSC MOC (ML19275D578)
  - "Collaboration on pre-application activities to ensure mutual preparedness to efficiently review advanced reactor and SMR designs"
  - A number of vendors proposing to use TRISO fuel are engaged in pre-licensing or licensing activities

- The TRISO assessment is a joint white-paper that can be used to develop regulatory guidance
  - Currently under final management review
  - Completion expected in the second quarter of 2023

Assessment Scope

- Considers recent TRISO fuel development work and existing guidance (e.g., NUREG-2246 (ML22063A131)) to:
  - Develop a shared, evidentiary basis to support regulatory findings for items that are generically applicable to TRISO
  - Identify items that are design dependent
  - Highlight areas where additional information or testing is needed

- Focused on the U. S. Department of Energy's (DOE's) Advanced Gas Reactor (AGR) program
  - Most applicants intend on using uranium oxycarbide (UCO) fuel kernels of the same or similar design

Assessment Scope

- Leverages the U.S. NRC staff's review of EPRI-AR-1-NP-A, UCO TRISO Coated Particle Fuel Performance topical report (TR) (ML20336A052)
  - TR scope included the AGR-1 and 2 test programs
  - TR focused on TRISO particle attributes that produced AGR program failure fractions and fission product releases

- The CNSC/NRC assessment will provide an overview of the following:
  - UCO TRISO particle assessment
  - Fuel compact or pebble form attributes
  - Evaluation model capabilities and model assessment
  - The AGR test envelope and the adequacy of AGR test data

NUREG-2246 Fuel Qualification for Advanced Reactors

NUREG-2246 is a technology inclusive framework that provides criteria that when satisfied support a regulatory finding that a nuclear fuel is qualified

- Qualified fuel refers to fuel that if built within specifications will perform as described in the safety analysis
- Primarily developed as a guide for advanced reactor fuel development since extensive guidance already exists for light-water reactor fuel
- Can be used by applicants to develop or assess existing fuel qualification plans or data
- Focused on solid fuel forms, NUREG/CR-7299 (ML22339A161) addresses fuel qualification for molten salt fueled reactors

UCO TRISO Particle and Fuel Form

- UCO particle attributes described in EPRI-AR-1-NP-A are sufficient to produce AGR fission product release performance
  - Qualitative standard for the SiC microstructure is subjective and can be difficult to implement
  - Work is currently being performed to characterize the AGR microstructure to better understand as-built grain size distribution
  - Data could be used to develop a quantitative standard for the SiC microstructure
- Fuel Form (Compact and Pebble) Assessment
  - Review is design specific
  - Need to provide data/testing to demonstrate safety functions are met
  - 40% upper bound packing fraction limit based on the AGR program

Evaluation Model Assessment

- Identifies important geometry, material, physical modeling considerations necessary to develop a TRISO evaluation model
  - Some failure modes may be excluded based on meeting the AGR manufacturing specifications precluding certain failure mechanisms
  - Some failure modes cannot be modeled based on the lack of sufficient data
    - Use experimental data to account for failure modes not modeled
    - Provide justification that the overall failure fraction is sufficiently conservative to account for the mechanisms not modeled

- Over the tested temperature ranges, there is likely sufficient AGR data to support model validation though the final justification of data sufficiency is the responsibility of the applicant
  - Design-specific evaluation models are anticipated

Test Envelope

Test envelope should be consistent with irradiation tests covering expected design-specific normal operation and transient conditions (i.e., the performance envelope)

- Maximum steady-state irradiated parameters per EPRI-AR-1-NP-A
- 1600 ^{o}C target peak anticipated operational occurrence (AOO) particle temperature
  - Based the low failure rate at 1600 ^{o}C during AGR safety testing
  - AOO peak particle temperature < 1600 ^{o}C could be warranted based on design-specifics
  - Applicant required to demonstrate that SARRDL and appropriate dose criteria or limits are met
  - Higher peak AOO TRISO particles temperatures could be justified
- 1700 ^{o}C target peak design basis accident (DBA) particle temperature
  - Based on AGR data showing an increase in failure rate from 1700 to 1800 ^{o}C
  - DBA peak particle temperature < 1700 ^{o}C could be warranted based on design-specifics
  - Applicant required to demonstrate the appropriate dose criteria or limits are met
  - Higher peak DBA TRISO particles temperatures could be justified

Test Envelope

- AGR safety testing did not include overpower transient testing such as rod withdrawal or rod ejection type reactivity insertions
  - Failure fractions assumed to be a function of absolute temperature, but rate of change could lead to other failure modes (e.g., melt, kernel swelling induced coating stresses)

- Based on NGNP project, transients ≥ 1 second have a negligible temperature change across the particle due to the thermal time constant
  - Short time constant allows for energy dissipation to the surrounding environment
  - Overpower transients ≥ 1 second expected to have a negligible increase in failure fractions as compared to other means (e.g., absolute temperature)
  - Overpower transients should still be evaluated based on the failure mechanisms associated with absolute temperature
  - For overpower transients < 1 second, additional justification is needed to demonstrate a non-conservative failure fraction is predicted

Test Envelope

- The quality of the AGR 1 and 2 test data (and hence TRISO particle development) judged to be of sufficient quality for licensing applications

- Experimental uncertainties in EPRI-AR-1-A, Section 6.5 provide acceptable measurement uncertainties for use in licensing applications

- AGR program test conditions constructed to match the expected operating condition of HTGRs with full scale TRISO particles
  - Test conditions match the expected operating conditions
  - No particle scale distortion
  - Distortions caused by compact or pebble geometry can be accommodated analytically if the matrix material is well characterized

Conclusions

This report establishes a common regulatory position on TRISO fuel qualification based on existing knowledge (e.g., AGR program) and identifies design-specific analytical or testing gaps that should be addressed to enable TRISO use in licensing applications.

- AGR program provided end-state attributes and established manufacturing specifications to produce fuel with fission product retention capabilities to support expected licensing applications
- The extent and quality of the AGR 1 and 2 data, both steady-state irradiation and safety testing, may be sufficient for evaluation model development over the range of conditions tested
- Additional test data, beyond the current AGR program safety test data, is not needed for overpower transients with durations ≥ 1 second
    - For overpower transients < 1 second, additional justification needed to address potential failure mechanisms based on a large temperature differential across the particle
- Fuel compact or pebble is expected to be design-specific and the applicant will be responsible for qualifying compact/pebble designs that meet their safety functions
    - 40% upper bound packing fraction established

Questions?

Questions for U.S. NRC:

jeffrey.schmidt2@nrc.gov; 301-415-4016

Questions for CNSC:

mediarelations-relationsmedias@cnsc-ccsn.gc.ca; 613-996-6860

Break

Meeting will resume at 2:35 pm EST

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 575 470 255#

NRC-CNSC MOC

Interim Joint Report on Classification of Structures, Systems and Components

Agenda

- Work Plan
- Scope of Safety Classification Project
- Interim Report Findings
  - Safety Classification Comparison and Effects
  - Pilot Design Rule Comparisons
- Engineering Design Rule Inputs
- Questions

Objectives of Work Plan

- Identify key similarities and differences in the safety significance determination process, the scope of SSCs subject to the process, and the process outcomes
- Identify key similarities and differences in the engineering design rules and specifications applied to each safety class and how this impacts the outcomes
- Review how each organization applies existing codes and standards and interacts with Standards Development Organizations (SDOs) to verify appropriate codes and standards are being developed, applied, and endorsed.

Scope of Work

- New Water-Cooled Small Modular and Advanced Non-Water-Cooled Reactors
- Safety Classification Processes:

Traditional NRC (Functional)	Licensing Modernization Project	CNSC Graded Approach
Safety Related	Safety Related	Important to Safety
Important to Safety/Not Safety Related (Includes PDC, DID, and RTNSS)	Not Safety Related with Special Treatment	
Not Important to Safety	No Special Treatment	Not Important to Safety

- Application of Engineering Design Rules:

Programmatic	Specific Design	Hazard Protection
Reliability Assurance (Design, Maintenance, and Availability)	Pressure Retaining Components	Seismic Design
Quality Assurance (Construction)	Civil Structures	Fire Protection
Testing and Inspection	Electrical and I&C	Equipment Qualification

Safety Classification Process

- Safety Analysis
  - Deterministic
  - Probabilistic
- Initiating Event Determination
- Safety Functions
- Consequence Assessment
- Classification of Structures, Systems, and Components (SSCs)
- Assignment of Engineering Design Rules by Classification

NRC Licensing Approach

- Addresses applications under 10 CFR Part 50 or 10 CFR Part 52
- 10 CFR 50.40, “Common Standards,” states: In issuing a construction permit or operating license under 10 CFR Part 50 or an early site permit, combined operating license, or manufacturing license under Part 52, the Commission will be guided, in part, by:
  - reasonable assurance of compliance with the regulations of 10 CFR Part 50
  - adequate protection of the public health and safety

NRC Safety Analysis Elements

- A safety assessment of the site and facility, including:
  - contained radioactive materials
  - application of engineering standards
  - safety features and barriers to release of radioactive material
  - analysis of a postulated fission product release
- An assessment of the design of the facility, including:
  - principal design criteria (PDC)
  - relationship of the facility design bases to the PDC
  - analysis and evaluation of the design and performance of SSCs to assess the risk to public health and safety

Definition of Safety-Related

- 10 CFR 50.2: *Safety-related SSCs* means those SSCs relied upon to remain functional during and following design basis events to assure:
  - The integrity of the reactor coolant pressure boundary;
  - The capability to shut down the reactor and maintain it in a safe shutdown condition; or
  - The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures.
- Influences requirements for traditional safety analysis and application of engineering design rules

NRC Traditional Approach

- Deterministic structure
  - Single failure criterion
  - Conservative analytical methods
  - Reliance on safety-related SSCs
  - Acceptance criteria related to initiating event frequency
- Design-specific probabilistic analyses provide risk insights and confirm safety goals would be met

Analysis Acceptance Criteria

Initiating Event Category	AOO	DBA
SSC Availability	Safety-Related SSCs with Single Failure; with and without Offsite Power; other SSCs with Technical Justification	Safety-Related SSCs with Single Failure; with and without Offsite Power
Pressure Boundary	Within 110% of Design	Within Acceptable Design Limits
Fuel	Within Specified Acceptable Fuel Design Limits	Cladding Failure if Specified Acceptable Fuel Design Limit Exceeded
Dose	10 CFR Part 20	Accident Dose Limit (25 Rem TEDE) or Small Fraction of Limit
Consequential Failures	No Escalation without other Independent Faults	No Consequential Failures of SSCs Necessary to Mitigate Fault
Loss of Coolant Accident	Not Applicable	10 CFR 50.46 Criteria

NRC Traditional Classification

- Safety-Related
  - SSCs relied on to meet analysis acceptance criteria for safe shutdown (including pressure boundary)
  - SSCs credited for mitigation of dose consequences
- Important to Safety
  - Functions identified in PDC
  - Special purpose regulations for defense in depth
  - Regulatory Treatment of Non-Safety Systems (RTNSS)
- Risk-informed safety classification per 10 CFR 50.69

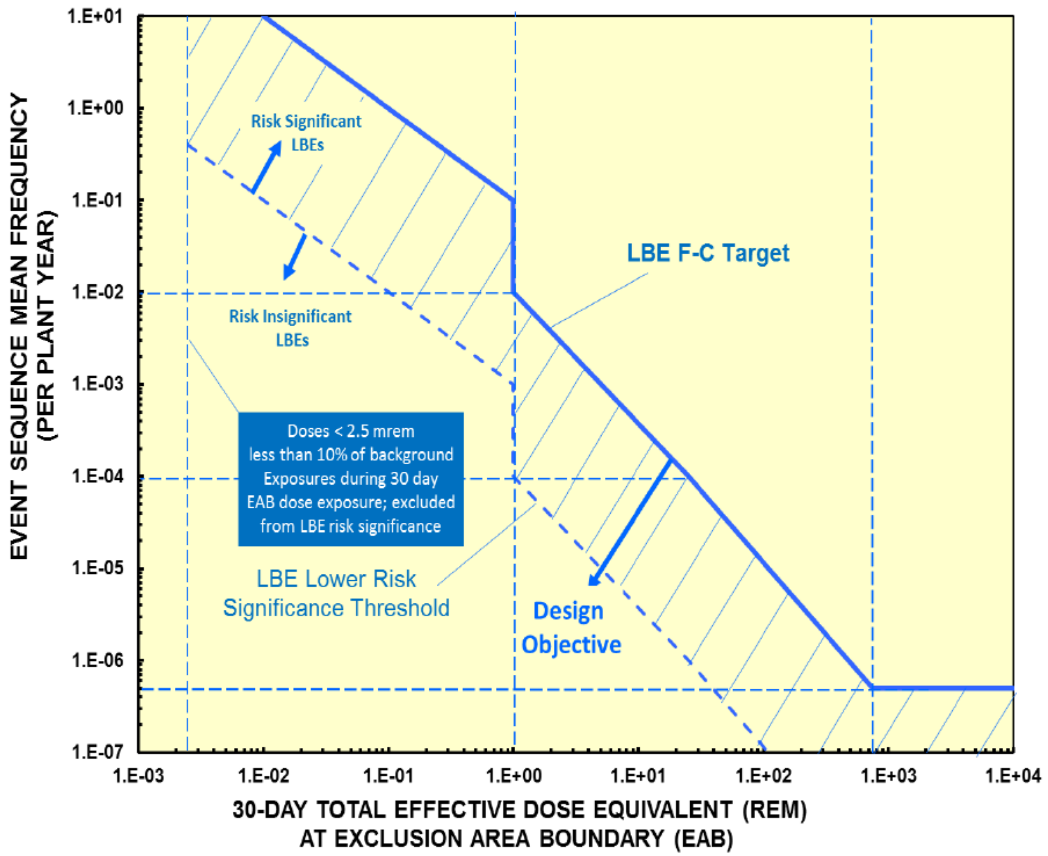
Engineering Design Rules

- NRC regulations associate application of certain rules based on SSC safety classification, for example:
  - Quality assurance for activities affecting the safety-related functions of SSCs
  - Seismic design criteria for safety-related SSCs
  - Inservice testing and inspection of safety-related SSCs (ASME Code per 10 CFR 50.55a, water-cooled reactors)
  - Environmental qualification of important to safety SSCs
- Other rules applied on a graded basis (GDC-1)

Licensing Modernization Project (LMP)

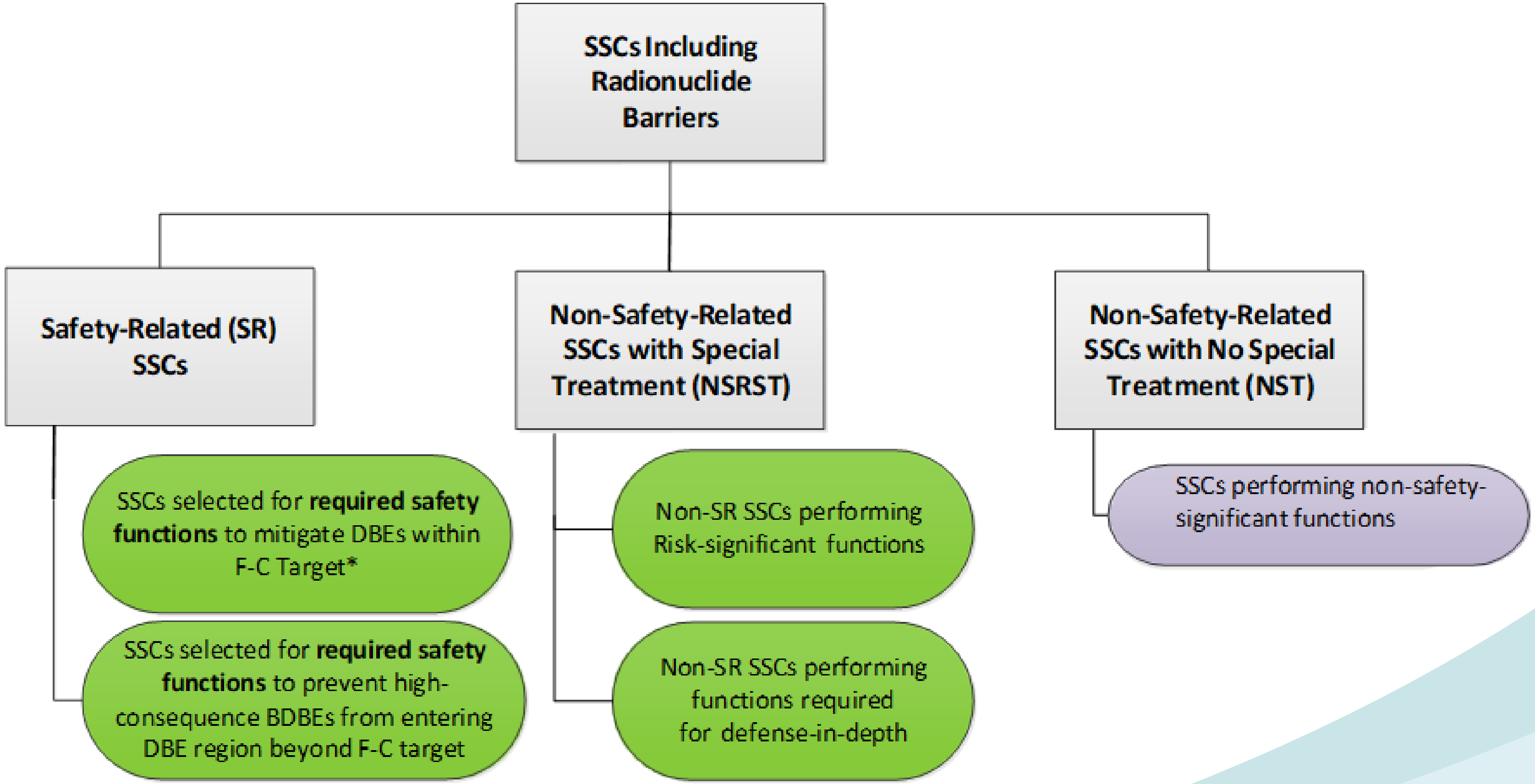
- Technology-inclusive, risk-informed, and performance-based licensing process
  - NEI 18-04 endorsed for licensing of advanced reactors within NRC regulatory framework (RG 1.233)
  - Establishes methods for the following:
    - Definition, categorization, and evaluation of events
    - SSC classification, performance criteria, and design rules
    - Evaluation of defense in depth adequacy
- Informs safety design to demonstrate compliance

LMP Event Classification



- Anticipated Operational Occurrences (AOOs) have frequencies of 10^{-2} per plant-year or higher
- Design basis events (DBEs) have frequencies of 10^{-4} or higher and less than 10^{-2} per plant-year
- Beyond design basis events (BDBEs) have frequencies less than 10^{-4} per plant-year
- Design basis accidents are event sequences derived from DBEs to set safety related SSC performance criteria

LMP Safety Classification



LMP Attributes

- Risk-informed with consideration of uncertainty
- Events evaluated with consideration of sequence frequency and consequences
- Criteria address cumulative and sequence risk
- Explicit consideration of defense in depth
- Assignment of engineering design rules considers safety classification and SSC safety function

NRC Exemption Process

- NRC regulations provide for specific exemptions that:
  - Are authorized by law
  - Will not present undue risk to public health and safety
  - Are consistent with the common defense and security
  - Supported by one or more special circumstances
- Special circumstances include:
  - Compliance not necessary to achieve the underlying purpose
  - Safety benefit compensates for any decrease in safety

CNSC Approach

- Nuclear Safety Control Act (NSCA) compliance required
- CNSC promulgates REGDOCs to meet NSCA
  - REGDOCs include requirements and guidance
  - Applicants may show intent of requirement has been addressed by other means; Commission determines if requirement is met
- Safety analysis expectations captured in:
  - REGDOC 2.4.1, “Deterministic Safety Analysis”
  - REGDOC 2.4.2, “Probabilistic Safety Assessment for Nuclear Power Plants”
  - REGDOC 2.5.2, “Design of Reactor Facilities: Nuclear Power Plants”

Analysis Acceptance Criteria

Event Category	AOO	DBA (or AOO with DID Level 2 Failure)	BDBA
SSC Availability	No Single Failure	Single-Failure Affecting Safety System Group	No Single Failure
Analysis Methods	Best Estimate (DID Level 2)	Conservative Analysis or Best Estimate plus Evaluation of Uncertainties (DID Level 3)	Best Estimate (DID Level 4)
Fuel and SSC Limits	Within Specified Acceptable Design Limits; No Unanalyzed Conditions	Within Specified Acceptable Design Limits; No Unanalyzed Conditions	Evaluate Ability to Restore or Maintain Safety Functions
Dose	0.5 millisievert (mSv)	20 mSv	Safety Goals
Consequential Failures	Prevented to the Extent Practicable	Prevented to the Extent Practicable	Avoid Cliff-Edge Effects; Prevent Early Containment Failure

Defence in Depth

- Defence in depth explicitly considered
- Five levels of defence:
  1. Prevent deviation from normal operation
  2. Prevent AOOs from escalating to accident conditions; control systems acting alone prevent SSC damage
  3. Minimize accident consequences; safety systems acting alone mitigate all AOOs and DBAs within dose criterion
  4. Minimize radiological release from severe accident; probabilistic analyses demonstrate safety goals are met
  5. Mitigate consequences of release

CNSC Safety Classification

- All SSCs identified as important to safety (ITS) or not important to safety
- Safety-significance of ITS SSCs based on:
  - safety function(s) to be performed
  - consequence(s) of failure
  - probability that the SSC will be called upon to perform the safety function
  - time following a initiating event at which the SSC will be called upon to operate, and the expected duration of that operation
- Applicant may propose graded classification of ITS SSCs

CNSC Assignment of Design Rules

- REGDOC 2.5.2 provides guidance for assigning engineering design rules
- Rules should be determined based on safety classification and include the following categories:
  - Codes and standards
  - Safety margins
  - Reliability
  - Equipment qualification
  - Provisions for inspection, testing, and maintenance
  - Organizational quality assurance

Safety Analysis Similarities

Design	Events	Analysis
Applicant proposes initial design and iterates to meet similar performance goals.	Identification of initiating events; categorization of events by frequency	Compare calculated consequences against performance criteria that vary with event frequency.

Safety Analysis Differences

Framework	CNSC	NRC Traditional	NRC LMP
Mitigating SSCs	Important to Safety SSCs	Safety Related (SR) only	SR only - Performance Criteria and Consequence Analysis
AOO Analyses	Sequence Frequency; Best Estimate	Initiating Frequency; Conservative Analysis	Sequence Frequency; Best Estimate w/Uncertainty
Accident Analyses	Sequence Frequency; Conservative or Best Estimate w/Uncertainty	Guidance for Event Selection; Conservative	Sequence Frequency; Best Estimate w/Uncertainty
Beyond Design Basis	Sequence Frequency; Best Estimate	Special Regulations; Best Estimate	Sequence Frequency; Best Estimate w/Uncertainty
Probabilistic Analyses	Complementary	Confirmatory	Foundational

Outcome of Safety Classification

<table>
<tr><th>Safety Significance</th><th colspan="4">High</th><th colspan="4">Low</th></tr>
<tr><td rowspan="2">CNSC</td><td colspan="6">Important to Safety (ITS)</td><td colspan="2">Not Important to Safety (NITS)</td></tr>
<tr><td>ITS - High</td><td colspan="3">ITS - Medium</td><td colspan="2">ITS - Low</td><td colspan="2">NITS</td></tr>
<tr><td rowspan="3">NRC LMP</td><td colspan="2">LMP Risk Significant
(and Safety Significant)</td><td colspan="3">LMP Safety Significant</td><td colspan="3">Not Safety Significant</td></tr>
<tr><td colspan="3">Safety-Related</td><td colspan="2"></td><td colspan="3" rowspan="2">Non-Safety-Related
No Special Treatment</td></tr>
<tr><td colspan="5">Non-Safety-Related with Special Treatment</td></tr>
<tr><td rowspan="3">NRC Traditional</td><td colspan="7">Important to Safety</td><td>NITS</td></tr>
<tr><td colspan="5">Safety-Related (RISC-1)</td><td colspan="2">Safety-Related (RISC-3)</td><td rowspan="2">NITS</td></tr>
<tr><td colspan="5">ITS (Not Safety-Related) (RISC-2)</td><td colspan="2">ITS (RISC-4)</td></tr>
</table>

Leveraging Prior Approvals

- Assumptions:
  - Identical single-reactor plant for deployment in U.S. and Canada
  - Applicant uses safety analysis method consistent with a selected regulatory framework
  - Applicant develops probabilistic analysis for confirmation of defense in depth and support of risk-informed decision-making

Leveraging Prior Approvals (Con't)

- Leveraging NRC Framework Outcome for CNSC Application
  - Conformance with CNSC regulatory requirements expected; risk-informed processes support justification of alternate means
  - Demonstrate conformance with defence in depth and engineering design rule assignment using risk informed processes

Leveraging Prior Approvals (Con't)

- Leveraging CNSC Framework Outcome for NRC Application
  - Development of principal design criteria (PDC), definition of SSCs considered equivalent to "safety-related", and application of design rules
  - Reconcile differences in safety analysis necessary to satisfy PDC (analysis of AOOs) and definition of "safety-related"
    - SSCs credited for mitigation
    - AOO categorization (initiating event or full sequence frequency) and acceptance criteria
  - Conformance with applicable special purpose regulations (exemption)
  - Address conformance with standard review plan for water-cooled reactors
  - Verify defense in depth

Design Rule Insights

- Reliability Assurance Programs
  - Establishes engineering design rules applied to intermediate safety-significance SSCs
  - Program consistent with risk-informed classification processes
- Pressure Retaining Components and Supports
  - Functional Classification (light water SMRs only):
    - Functional classification results in the application of ASME BPVC Section III, Division 1
    - Differences in functional classification increase for lower safety-significance SSCs
  - Risk informed, technology neutral classification guidance likely to support consistent application of codes to individual SSCs (SMRs and Advanced Reactors)

Next Steps

- Finalize engineering design rules topic area input addressing similarities, differences, and impacts

Programmatic	Specific Design	Hazard Protection
Reliability Assurance (Design, Maintenance, and Availability)	Pressure Retaining Components	Seismic Design
Quality Assurance (Construction)	Civil Structures	Fire Protection
Testing and Inspection	Electrical and I&C	Equipment Qualification

- Expected release of final report in Summer 2023

THANK YOU

Questions?

Future Meeting Planning

- The next periodic stakeholder meetings are scheduled for the following dates in 2023: June 7, July 20, and September 14.

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

How Did We Do?

- Click link to NRC public meeting information:

  https://www.nrc.gov/pmns/mtg?do=details&Code=20230268

- Then, click link to NRC public feedback form:

Meeting Feedback

Meeting Feedback Form EXIT

Meeting Dates and Times