Advanced Reactor Stakeholder Public Meeting

July 20, 2023

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 501 432 683#

Time	Agenda	Speaker
10:00 am – 10:10 am	Opening Remarks / Advanced Reactor Integrated Schedule	NRC
10:10 am – 10:45 am	Environmental Center of Expertise Licensing Review Overview and Enhancements	NRC
10:45 am – 11:00 am	Introduction to the New Fuels Atlas	NRC
11:00 pm – 11:30 am	Update on SCALE/MELCOR Non-LWR Source Term and Fuel Cycle Demonstration Project	NRC
11:30 am – 12:00 pm	Nuclear Supplier QA Program Qualification: ISO 9001 Supplemental Requirements	Nuclear Energy Institute (NEI)
12:00 pm – 1:00 pm	Lunch Break	All
1:00 pm – 1:30 pm	Insights on Role of Advisory Committee on Reactor Safeguards (ACRS) During Initial Licensing Reviews	NRC
1:30 pm – 2:30 pm	Advanced Reactor Population-Related Siting Considerations	NEI
2:30 pm – 2:45 pm	Break	All

Time	Agenda (continued)	Speaker
2:45 pm – 3:15 pm	MACCS Consequence Analysis Demonstration Calculations	NRC
3:15 pm – 4:15 pm	Factory-Fabricated Transportable Micro-Reactor Licensing and Deployment Considerations	NRC
4:15 pm – 4:30 pm	Break	All
4:30 pm – 5:00 pm	Risk-Informed Approach to Package Approval for Transportable Microreactors	NRC
5:00 pm – 5:05 pm	Future Meeting Planning and Concluding Remarks	NRC

Availability of Interim Joint Report on: Classification of Structures, Systems, and Components

- Interim report available through link on: https://www.nrc.gov/reactors/new-reactors/advanced/international-cooperation/collaboration-with-canada.html
- Also available in NRC ADAMS at ML23172A201
- Feedback is welcome, especially related to enhancing usefulness for potential license applicants
- Final report expected to be issued this fall

Contact: Steve.Jones@nrc.gov or Jorge.Hernandez@nrc.gov

NRC's Advanced Reactor Readiness By the Numbers

Statistics since 2018

Work on more than **35 policy issues** created more than **60 guidance documents**.

10 NRC/DOE MOUs focused on advanced reactor collaboration.

Canada collaboration generated more than **10 work plans**, **9 NRC/CNSC joint reports.**

Completed more than **10 advanced reactor design** reference models to make future assessments more efficient.

Established core review teams of **8-10 technical staff** per application, based on recent new reactor review experience.

More than **140 public engagements** per year on advanced reactor-related topics

Completed more than **75 topical report/white paper reviews** **33% faster** than the generic schedule goal.

Completed Kairos construction permit safety review **50% faster** than the generic schedule goal.

***The NRC's strategic transformation and modernization enables the safe deployment of* ADVANCED REACTORS**

Advanced Reactor Integrated Schedule of Activities

The updated Advanced Reactor Integrated Schedule

is publicly available on NRC Advanced Reactors website at:

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

Advanced Reactor Integrated Schedule of Activities

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

Legend

Concurrence (Division/Interoffice)	• 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 7/11/23

Present Day

Strategy	*Regulatory Activity*	*Commission Papers*	*Guidance*	*Rulemaking*	*NEIMA*	*Complete*	2022 Jan	2022 Feb	2022 Mar	2022 Apr	2022 May	2022 Jun	2022 Jul	2022 Aug	2022 Sep	2022 Oct	2022 Nov	2022 Dec	2023 Jan	2023 Feb	2023 Mar	2023 Apr	2023 May	2023 Jun	2023 Jul	2023 Aug	2023 Sep	2023 Oct	2023 Nov	2023 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																													
	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 (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					x																								
	Non-LWR MELCOR (Source Term) Demonstration Project					x									↓															
	Reference SCALE/MELCOR plant model for					x																								

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

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Advanced Reactor Program

Environmental Center of Expertise Licensing Review Overview and Enhancements

Kenneth Erwin
Office of Nuclear Material Safety and Safeguards
Division of Rulemaking, Environmental, and Financial Support
Environmental Center of Expertise

Advanced Reactor Stakeholder Meeting
July 20, 2023

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Principle Legislation, Regulations, and Outcome

Environmental Review

Impacts **ON** the environment **FROM** the facility

- National Environmental Policy Act (NEPA) (1969)
- National Historic Preservation Act (1966), Endangered Species Act (1973), others
- 10 CFR Part 51
- Impact-focused analyses
- Disclosure document (CatEx, EA/FONSI, EIS/ROD)

Safety Review

Impacts **ON** the facility **FROM** the environment

- Atomic Energy Act (1954)
- Energy Reorganization Act of (1974)
- 10 CFR Parts as applicable (e.g., 20, 40, 50, 52, 70, 71, 72, 100)
- Risk Informed
- Reasonable assurance of adequate protection (SER)

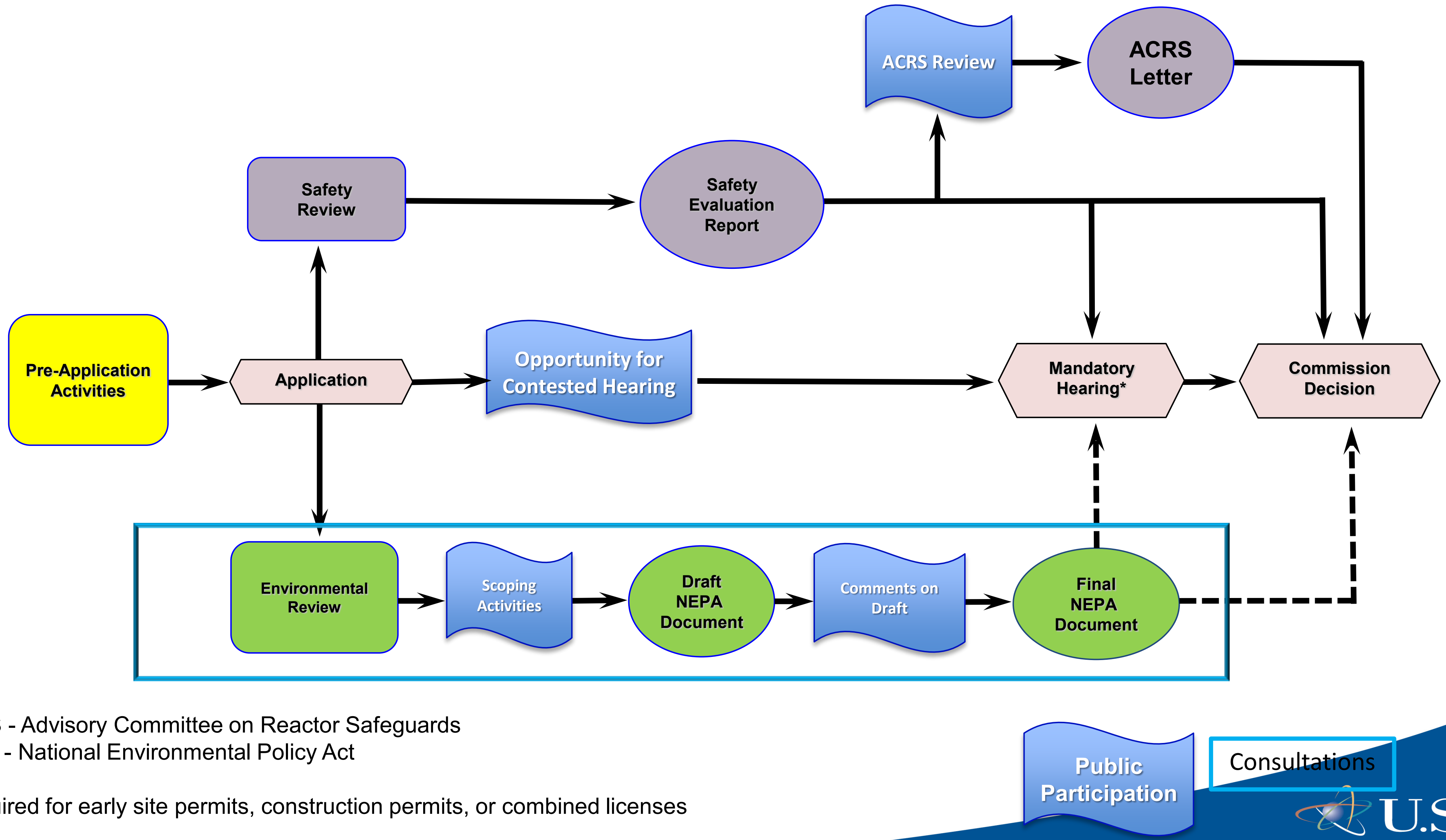
Other Important Statutes

- National Waste Policy Act (1982)
- Energy Policy Act (2005)
- Nuclear Energy Innovation and Modernization Act (amended 2019)
- Title 41 of Fixing America's Surface Transportation Act (FAST-41) (2015)
- Fiscal Responsibility Act (2023)

CatEx – Category Exclusion
EA – Environmental Assessment
FONSI – Finding of No Significant Impact
EIS – Environmental Impact Statement
ROD – Record of Decision
SER – Safety Evaluation Report

9

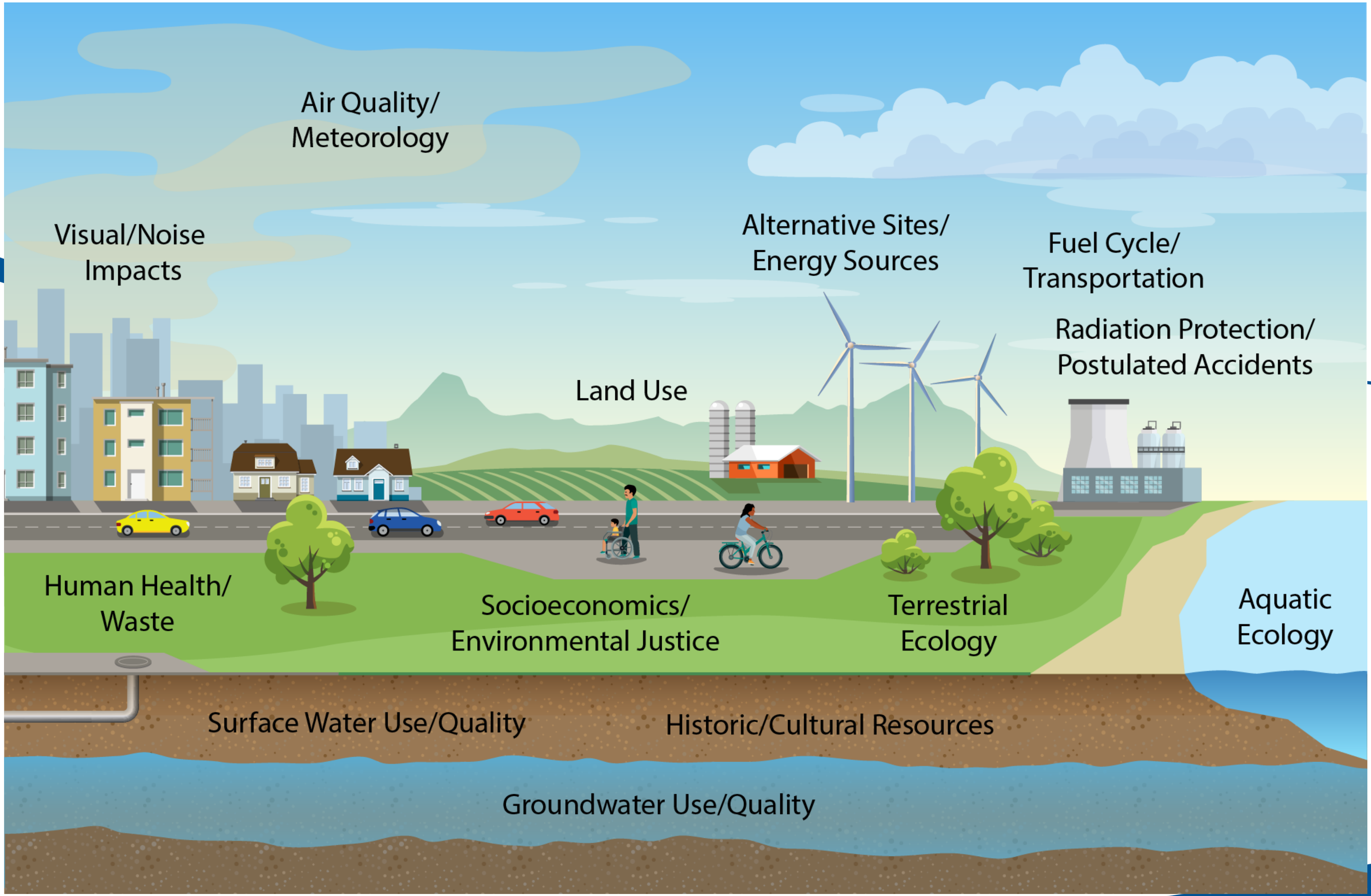
Licensing Process and the Environmental Review



ACRS - Advisory Committee on Reactor Safeguards
NEPA - National Environmental Policy Act

*Required for early site permits, construction permits, or combined licenses

Resource Areas Analyzed in NRC's NEPA Reviews*



Describe the "affected environment" (baseline conditions) for each resource area and then the consequences of the action (impact level).

Analyze "cumulative impacts" from past, present, or reasonably foreseeable future actions.

*Typical resources analyzed in new reactor NEPA reviews

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Coordination with Federal, State, local, and Tribal Government Agencies

United States Environmental Protection Agency

Department of Energy · United States of America

US Army Corps of Engineers®

Federal Permitting Improvement Steering Council · United States of America

National Oceanic and Atmospheric Administration · NOAA · U.S. Department of Commerce

U.S. Fish & Wildlife Service · Department of the Interior

Forest Service · US · Department of Agriculture

U.S. AIR FORCE

United States Nuclear Regulatory Commission

National Park Service

Advisory Council on Historic Preservation · ACHP · 1966

National Association of Tribal Historic Preservation Officers

NCSHPO
National Conference of State Historic Preservation Officers

NRC Environmental Center of Expertise (ECOE)

- **Genesis:** Established in October 2019.
- **Organization:** Centralized Branches From Different Offices Into One Division in the Office Material Safety and Safeguards (NMSS)
- **Mission:** Performs NRC-wide NEPA Reviews
  - Streamlined processes, procedures, and guidance
  - Developed common skillsets, knowledge management, project tracking
  - Created NEPA "Toolbox," Project Management Handbook, and internal NRC web pages (Nuclepedia)
- **Strategic Plan:** Implement NRC's Mission and Vision (Link)
  - Ensure the safe and secure use of radioactive materials
  - Continue to foster a healthy organization
  - Inspire stakeholder confidence in the NRC

NRC Environmental Center of Expertise (ECOE)

- **Transformation:** Improve the efficiency and effectiveness of NRC's environmental review process
- Implemented Formal Internal Self Assessments and Transformation Efforts
  - Streamlined reviews and focus more on impacts that matter
  - Solicited input from staff, management, business line owners, stakeholders
  - Ensured NRC's NEPA obligations are met and defensible

Transformation Efforts

- Manage Projects using commercial project management software and apply agile project management techniques for workload optimization
  – Track project schedules, skillsets, and priority across all business lines in NRC (>200 projects)
  – Allows for prioritization, flexibility, and agility
  – Ensure targets and goals are met

- Incorporation of Lessons Learned
  – Clinch River Early Site Permit (ML19190A078)
  – Feedback from Public Meetings (ANR Stakeholder, Scoping, Draft EIS meetings)
  – Regulatory Information Conference (RIC) Sessions in 2020 (RIC 2020), 2021 (RIC 2021), and 2023 (RIC 2023 Environmental and RIC 2023 Siting)
  – Various stakeholder input (ML20065N155) and responses (ML20147A540) and (ML20183A475)
  – Regulatory Guide 4.2 Update and Feedback (2018) (ML18071A400)
  – Internal Review of New Reactor Reviews Lessons Learned Report for Environmental Reviews conducted in 2017

Transformation Outcomes

- Developed Advanced Nuclear Reactor (ANR) Generic Environmental Impact Statements (GEIS) (ML21222A044)
  - Technology Neutral, Performance Based (Site and Plant Parameter Envelopes) Framework
  - Use of bounding analysis and related concepts
  - Under review by the Commission
- Update to License Renewal Generic Environmental Impact Statements (GEIS) (FRN)
  - Incorporated lessons learned from previous License Renewal reviews
  - Accelerated schedule per Commission direction
- Improved NEPA documentation
  - Developed Templates
  - Improved Readability/Reduction in Redundancy
  - Increased use of Incorporation by Reference

Transformation Outcomes

- Continued Full Participation in Congressional and Administration Efforts and Situational Awareness, As Appropriate
  - Nuclear Energy Innovation Capabilities Act (Law)
  - Nuclear Energy Innovation and Modernization Act (Law)
  - Advance Act (Proposal)
  - FRA (Law)
  - FAST-41 Implementation through Federal Permitting Improvement Steering Council (FPISC) (Administration Website)
  - Council on Environmental Quality (CEQ) Guidance Updates (FRN)
  - CEQ Regulatory Changes (CEQ Website)
  - Executive Orders

- 10 CFR Part 51 SECY Paper and Alternatives Analysis Concepts Therein (ML20212L393)
- Initiated Update to Environmental Standard Review Plan (January 2023)
- Preparation of Brownfield Paper (November 2022)
- Developed MOUs with Other NEPA Responsible Agencies (i.e., Department of Energy)

Enhancing Advanced Reactor Reviews

- Robust Pre-application Engagement and Readiness Assessments
  - Support and Implement Business Line Owner's Efforts
    - 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 Explain NEPA Requirements and Support Readiness Assessments and use of NRR Office Instruction LIC 116
- Expanded Use of Public Meetings and Regulatory Audits
  - NMSS follows NRR Office Instruction LIC-111
  - Optimization based on lessons learned
- Optimized use of Requests for Additional Information (RAIs)
  - NMSS follows NRR Office Instruction LIC-115
  - Management review of RAIs before issuance
  - Coordination with business line owner
- Transparency through use of Dashboards

Enhancing Staff Capability and Capacity

- Multidisciplinary core review teams to focus reviews, as appropriate
- Formal Qualification Program for Project Managers and Technical Reviewers
  - Building capacity for multiple ongoing reviews
  - Hiring new staff
  - Training staff on advanced reactor, fuel cycle, and license renewal technology
  - Use of contractors for flexibility and agility
- Pre-application engagement with staff regarding site-specific NEPA resource technical review areas will support efficient reviews
- Timely information on industry plans supports effective NRC resource planning

Successfully Implementing Enhancements

- Kairos Hermes Test Reactor Construction Permit (CP) review
  - Draft EIS (Link) issued September 29, 2022, Final EIS expected September 2023; With Significant Improvements to Transparency, Accountability, and Readability
  - Supporting 21-month review schedule
    - Dashboards
    - Audits
    - Internal project controls
    - Multidisciplinary core review team
- Abilene Christian University Molten Salt Research Reactor CP review
  - Innovative use of Environmental Assessment (FRN)
- Pre-application reviews ongoing with multiple developers/site owners
  - Regulatory Engagement Plans
  - Successful completion of Pre-application Readiness Assessments, Public Meetings, and Site Visits
  - Pre-application assessments enhance readiness and quality of applications

Successfully Implementing Enhancements

- Support for Rulemakings and Policy Issues
  - 10 CFR Part 53 EA (FRN)
  - Catex (FRN)
  - Fusion (Website)
  - Accident Tolerant Fuel (Website)

Next Steps

- Continue stakeholder engagement through our pre-application readiness assessments/engagements and periodic advanced reactor public meetings
  – Share best practices with prospective applicants
- Continue to make enhancements to internal processes based on lessons learned from ongoing reviews and stakeholder input
  – Continue to assess our review processes during ongoing reviews
- Examine use of programmatic or sitewide EISs

Abbreviations/Acronyms

10 CFR – Title 10 of the *Federal Codes of Regulations*

ACRS – Advisory Committee on Reactor Safeguards

ANR – Advanced Nuclear Reactor

BL – Business Line

CatEx – Category Exclusion

CEQ – Council on Environmental Quality

EA – Environmental Assessment

ECOE – Environmental Center of Expertise

EIS – Environmental Impact Statement

FAST-41 - Title 41 of Fixing America's Surface Transportation Act

FONSI – Finding of No Significant Impact

FRN – *Federal Register* Notice

GEIS – Generic Environmental Impact Statement

MOU – Memorandum of Understanding

NRC – Nuclear Regulatory Commission

NEPA – National Environmental Policy Act

NMSS – Office of Nuclear Material Safety and Safeguards

RAI – Request for additional information

RIC – Regulatory Information Conference

ROD – Record of Decision

SLR – Subsequent License Renewal

Introduction to the New Fuels Atlas

Chris Markley

NMSS/DFM

Advanced Reactor Stakeholder Meeting

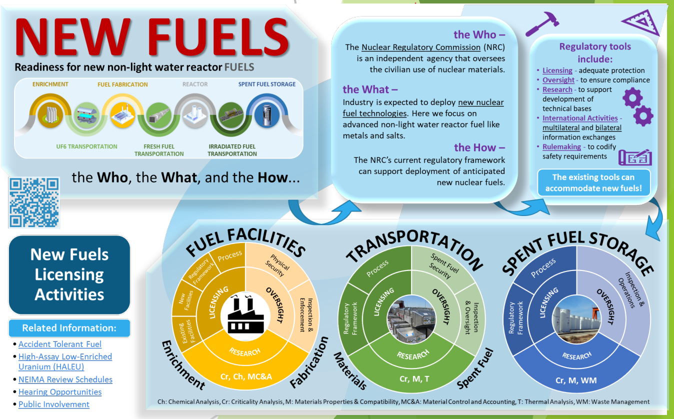
July 20, 2023

New Fuels Environment

- New fuels arena is evolving quickly
- Purpose: Enhance ability to identify and process information
- Outcome: New Fuels Atlas
  - Enhanced communications
    - Infographic
    - New Fuels Website
  - Enhanced organization
    - Regulatory Planner

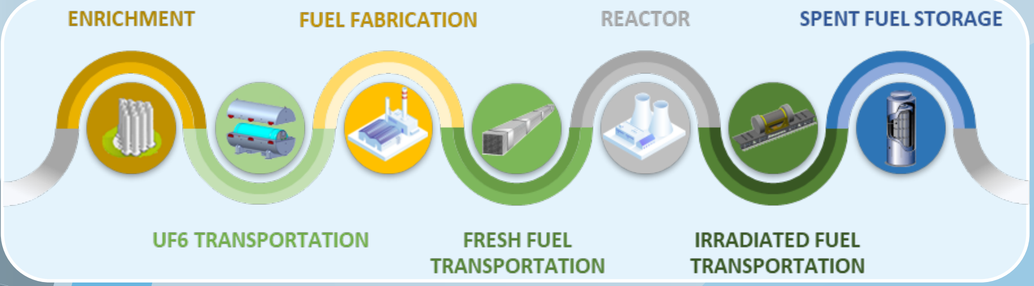
New Fuels Infographic

- Looks at all phases of the front and back end of the fuel cycle
- Provides the who, the what, and the how
- Highlights information for public stakeholders
  - Framework supports current environment
  - NRC has tools available to regulate



NEW FUELS

Readiness for new non-light water reactor FUELS



the **Who**, the **What**, and the **How**...

the Who –

The Nuclear Regulatory Commission (NRC) is an independent agency that oversees the civilian use of nuclear materials.

the What –

Industry is expected to deploy new nuclear fuel technologies. Here we focus on advanced non-light water reactor fuel like metals and salts.

the How –

The NRC's current regulatory framework can support deployment of anticipated new nuclear fuels.

Regulatory tools include:

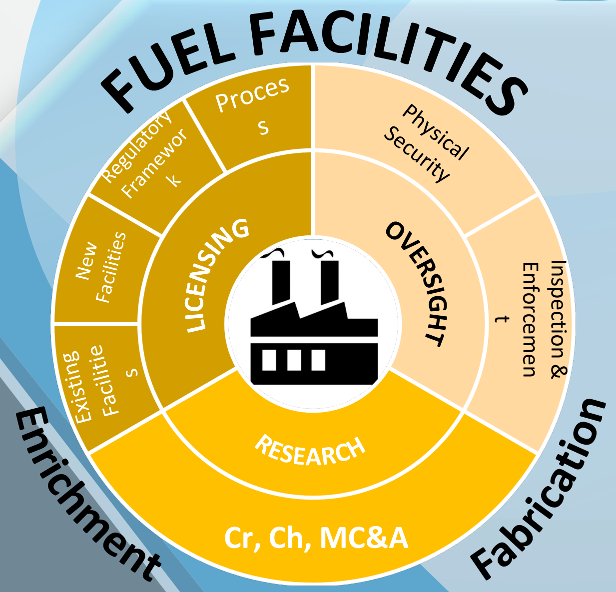
- **Licensing** - adequate protection
- **Oversight** - to ensure compliance
- **Research** - to support development of technical bases
- **International Activities** - multilateral and bilateral information exchanges
- **Rulemaking** - to codify safety requirements

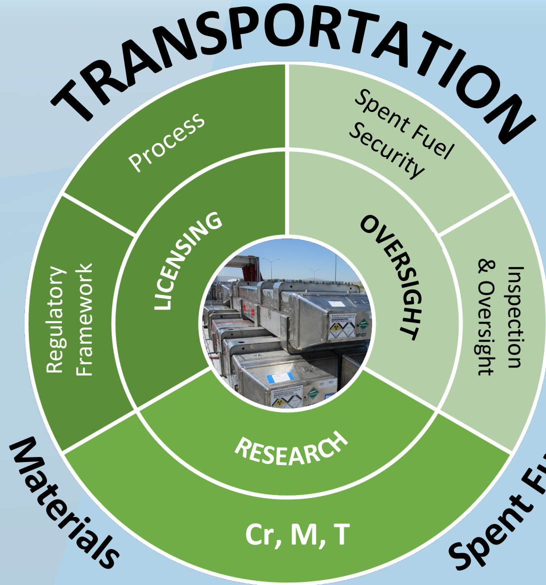
The existing tools can accommodate new fuels!

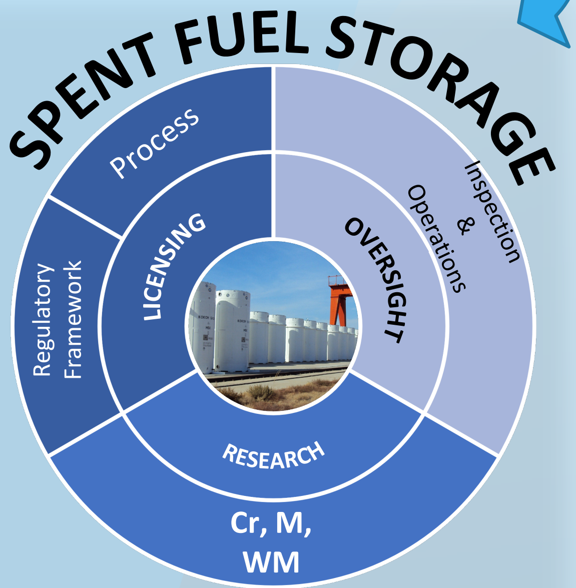
New Fuels Licensing Activities

Related Information:

- Accident Tolerant Fuel
- High-Assay Low-Enriched Uranium (HALEU)
- NEIMA Review Schedules
- Hearing Opportunities
- Public Involvement



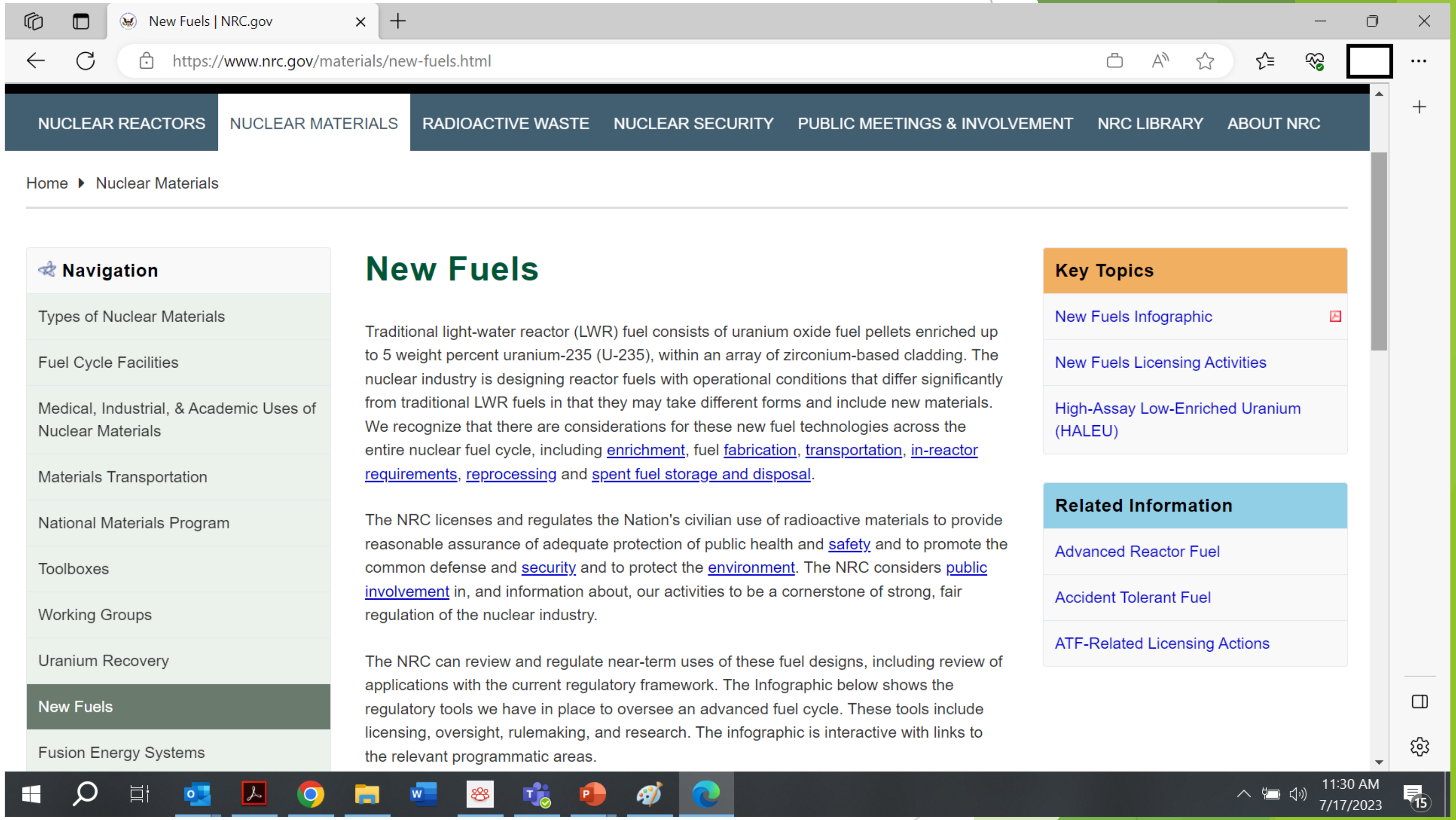




Ch: Chemical Analysis, Cr: Criticality Analysis, M: Materials Properties & Compatibility, MC&A: Material Control and Accounting, T: Thermal Analysis, WM: Waste Management

New Fuels Website

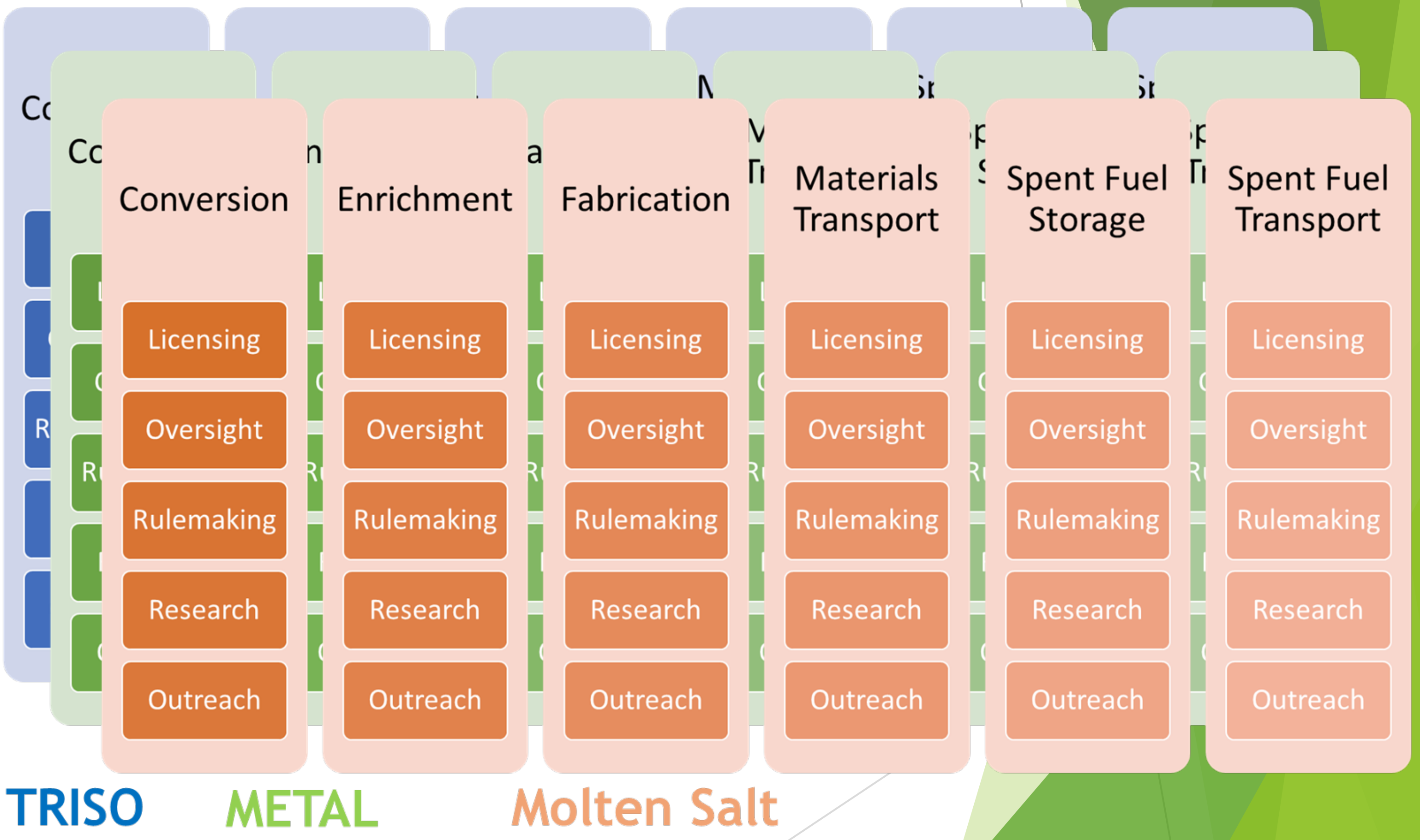
- Enrichment
- Fabrication
- Transportation
- Utilization
- Safety
- Environmental Protection
- Security and Safeguards
- Stakeholder Engagement



https://www.nrc.gov/materials/new-fuels.html

The Regulatory Planner

- Organizational tool
- For each technology
  - Fuel cycle phase
  - Programmatic area



Any Questions?

Update on SCALE / MELCOR non-LWR Source Term and Fuel Cycle Demonstration Project

Advanced Reactor Stakeholder Meeting
July 2023

Lucas Kyriazidis & Shawn Campbell
Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel & Source Term Code Development Branch

NRC's Strategy for Preparing for non-LWR Applications

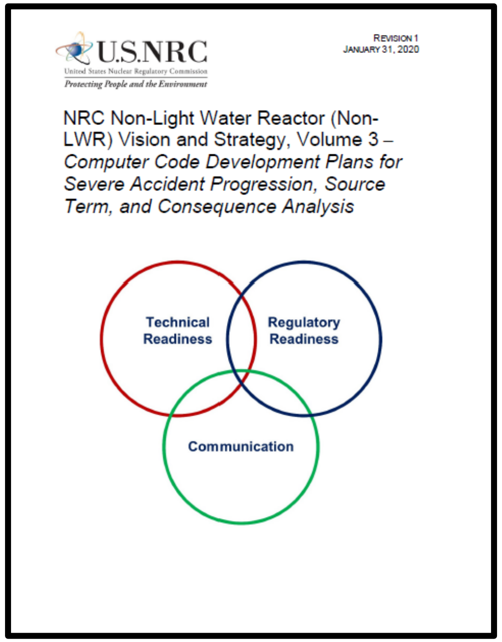
- *NRC's Readiness Strategy for Non-LWRs*

Strategy #1 Technical Skills	**Strategy #2** Computer Codes & Tools	**Strategy #3** Review Processes
Strategy #4 Codes & Standards	**Strategy #5** Policy Issues	**Strategy #6** Communications

- *IAPs are planning tools that describe:*
  - *Work, resources, and sequencing of work to achieve readiness*

- *Strategy #2 – Computer Codes and Review Tools*
  - *Identifies computer code & development activities*
  - *Identifies key phenomena*
  - *Assess available experimental data & needs*

NRC's Non-LWR Demonstration Projects & Codes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REVISION 1
JANUARY 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

Volume #3

OAK RIDGE National Laboratory

scale

NRC's comprehensive neutronics package

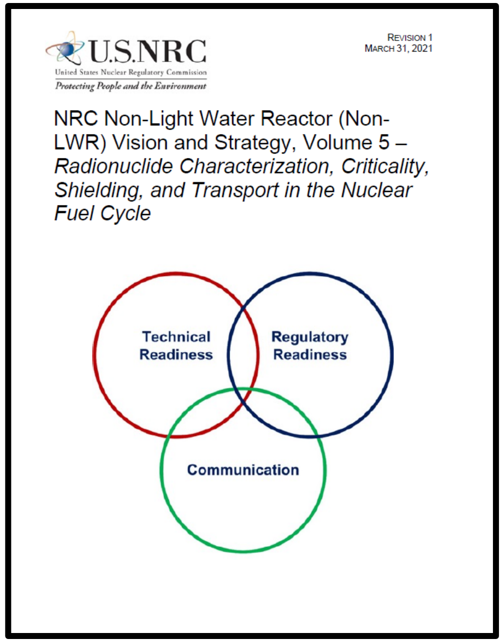
- Cross-section processing
- Decay heat analyses
- Criticality safety
- Radiation shielding
- Radionuclide inventory & depletion generation
- Reactor core physics

Sandia National Laboratories

MELCOR

NRC's comprehensive severe accident progression and source term code

- Accident progression
- Thermal-hydraulic response
- Core heat-up, degradation, and relocation
- Fission product release and transport behavior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REVISION 1
MARCH 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

Volume #5

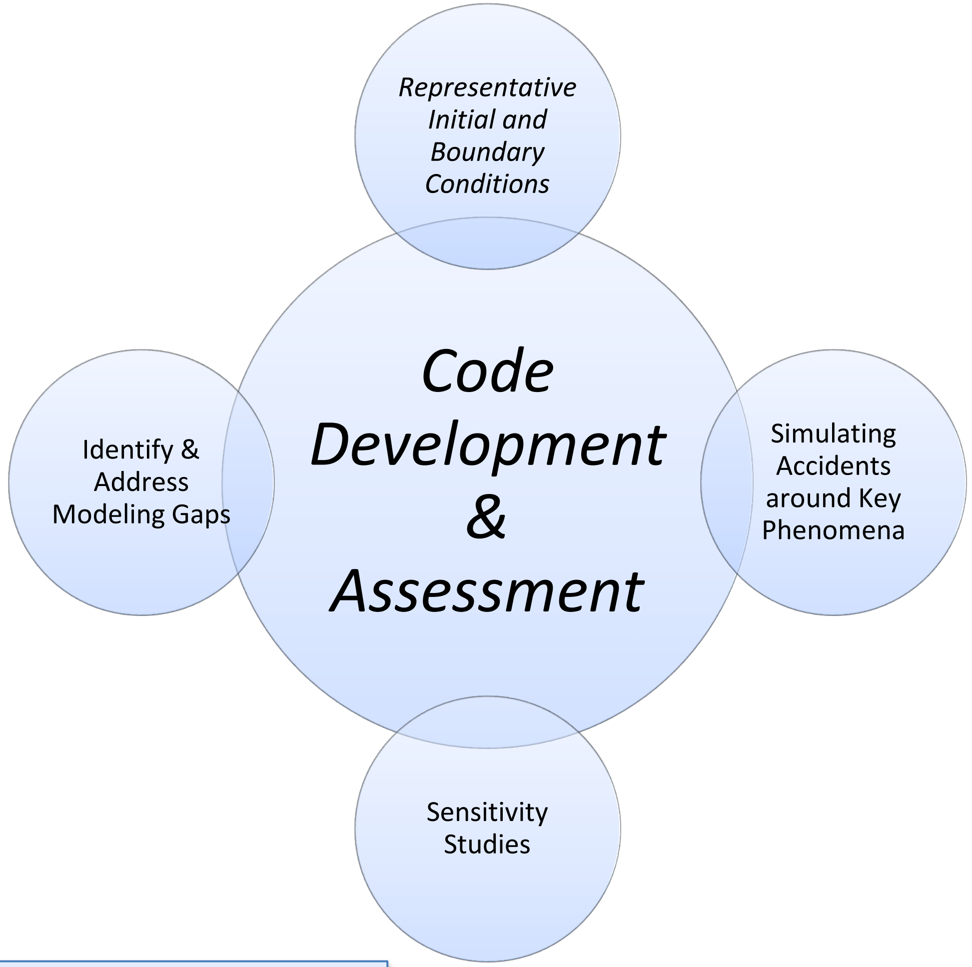
Goal of Volume 3 & 5 is demonstration of SCALE & MELCOR for simulating non-LWRs

General Approach

1. Build SCALE core models and MELCOR full-plant models

2. Select scenarios that demonstrate code capabilities for key phenomena

3. Perform simulations & code assessments on SCALE & MELCOR



- The scenarios and design assumptions were chosen to show capabilities of the new modeling features added to the codes.
- There is no significance to the magnitude of the releases in the MELCOR demonstration calculations.
- The results are not intended to provide accident source terms for use in licensing decisions.

Volume 3

Severe Accident Progression & Source Term

Availability of Volume 3 Reference Material

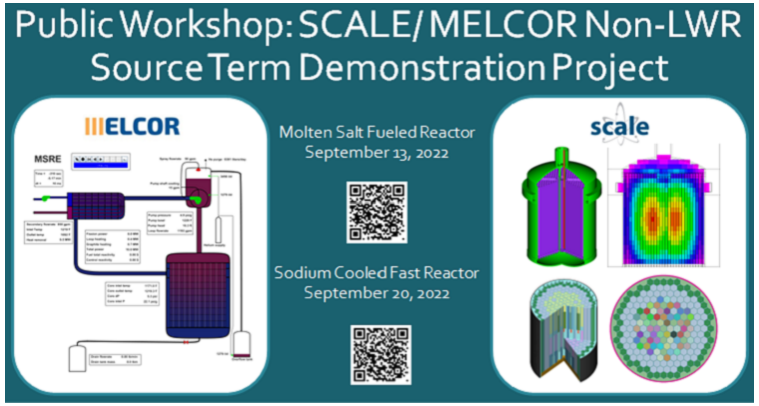
Five Major Types of Non-LWRs Analyzed under Volume 3

2021

- Heat Pipe Reactor – INL Design A
- High-Temperature Gas-Cooled Pebble-bed Reactor – PBMR-400
- Molten-salt-cooled Pebble-bed Reactor – UCB Mark 1

2022

- Molten-salt-fueled Reactor – MSRE
- Sodium-cooled Fast Reactor – ABTR



SCALE/MELCOR non-LWR source term demonstration project	

• Heat-pipe reactor workshop on June 29, 2021
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	June 29, 2021

• High-temperature gas-cooled reactor workshop on July 20, 2021
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	July 20, 2021

• Fluoride-salt-cooled high-temperature reactor workshop on September 14, 2021
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 14, 2021

• Molten-salt-fueled reactor workshop on September 13, 2022
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
	September 13, 2022

• Sodium-cooled fast reactor workshop on September 20, 2022
  ◦ Slides
  ◦ Video Recording EXIT
	September 20, 2022

Public workshop videos, slides, reports at advanced reactor source term webpage

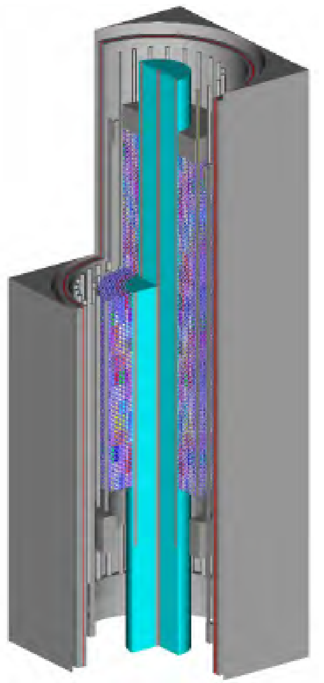
SCALE input models available here.

MELCOR input models available upon request.

Non-LWR Designs Considered & Project Scope

U.S.NRC
Oak Ridge National Laboratory
Sandia National Laboratories

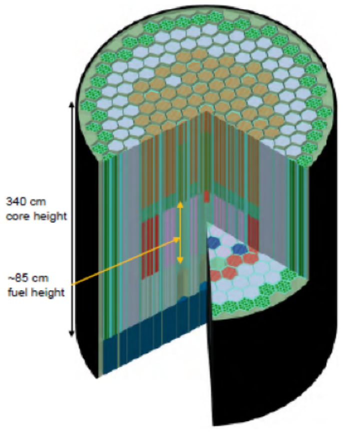
High-Temp. Gas Cooled Reactor



PBMR-400

- 400 MWth reactor, graphite moderated
- Helium-cooled & TRISO-particle pebble-fueled at 10 wt.% U-235
- Fuel discharged at high burnup (90 GWd/MTIHM)

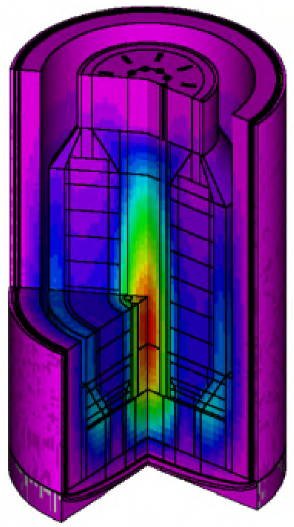
Sodium-Cooled Fast Reactor



ABTR

- 250 MWth pool-type reactor, utilizing metallic U / HT-9 fuel rods
- Reactor fueled with U-Pu-Zr fuel slugs
- Liquid sodium coolant

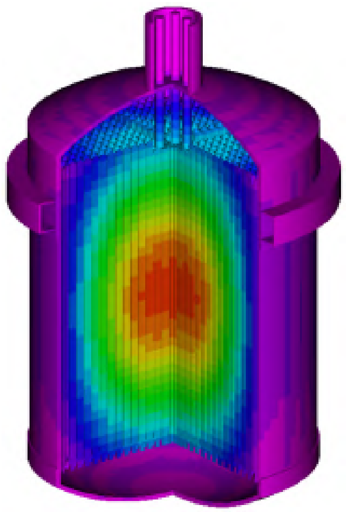
Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling

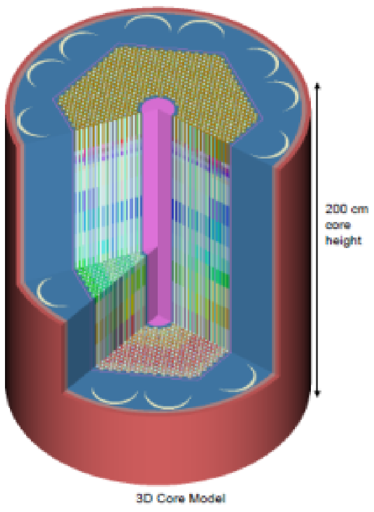
Molten Salt-Fueled Reactor



MSRE

- 10 MWth reactor, graphite moderated at near atmospheric pressures
- Reactor fueled with liquid dissolved fuel in molten salt (34.5 wt. % U-235)

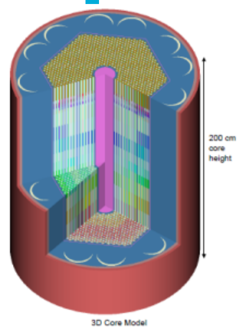
Heat Pipe Reactor

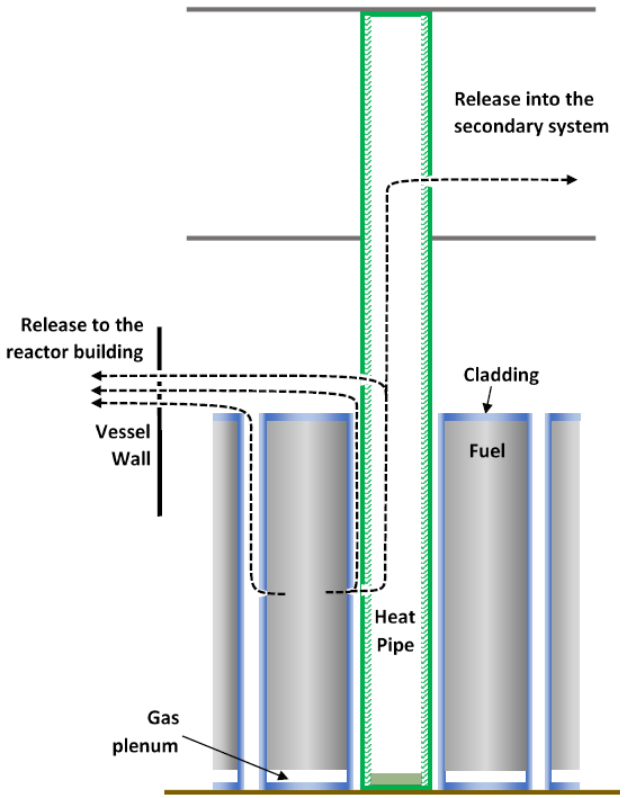


INL Design A

- 5 MWth with a 5-year operating lifetime
- 1,134 heat pipes fueled with UO_2 fuel (19.75 wt.% U-235)
- Reactivity controlled via control drums

Heat Pipe Reactor – INL Design A



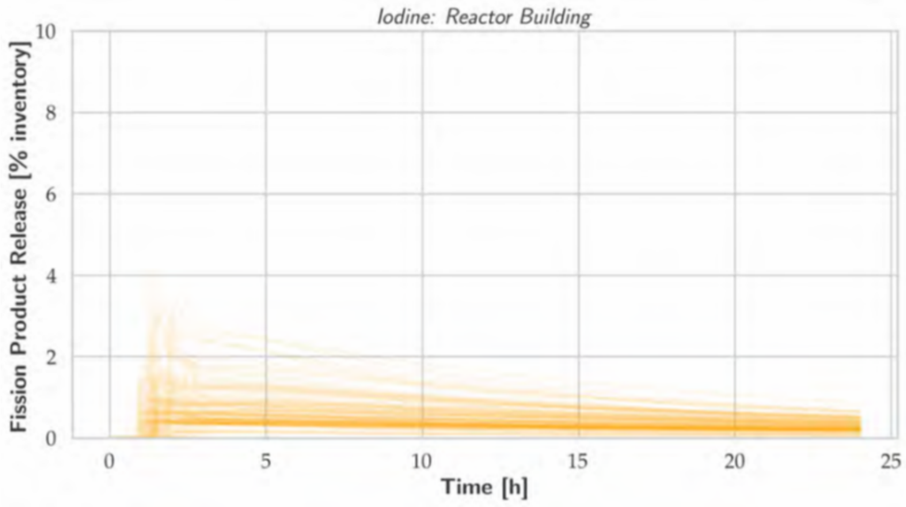


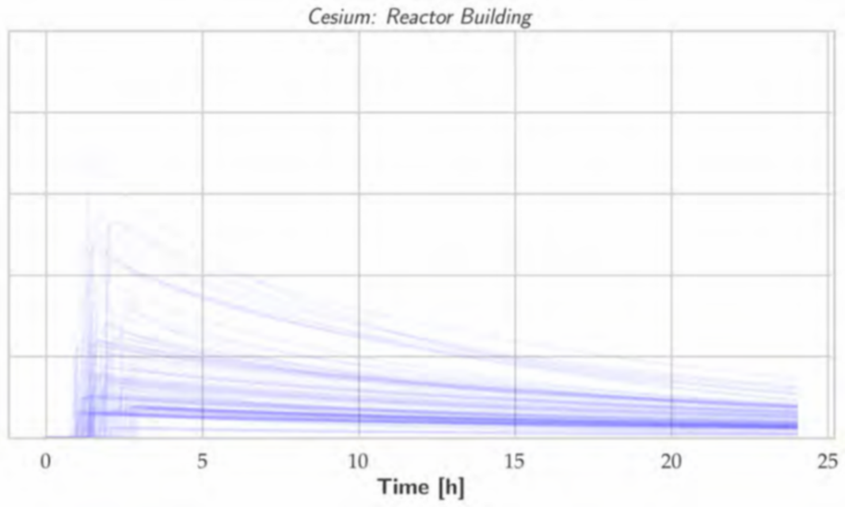
Accidents Modeled

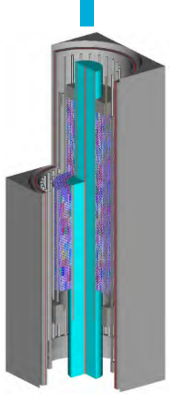
- Transient overpower
- Loss-of-heat sink
- Anticipated transient w/o SCRAM

Insights Gained on BDBA Behavior

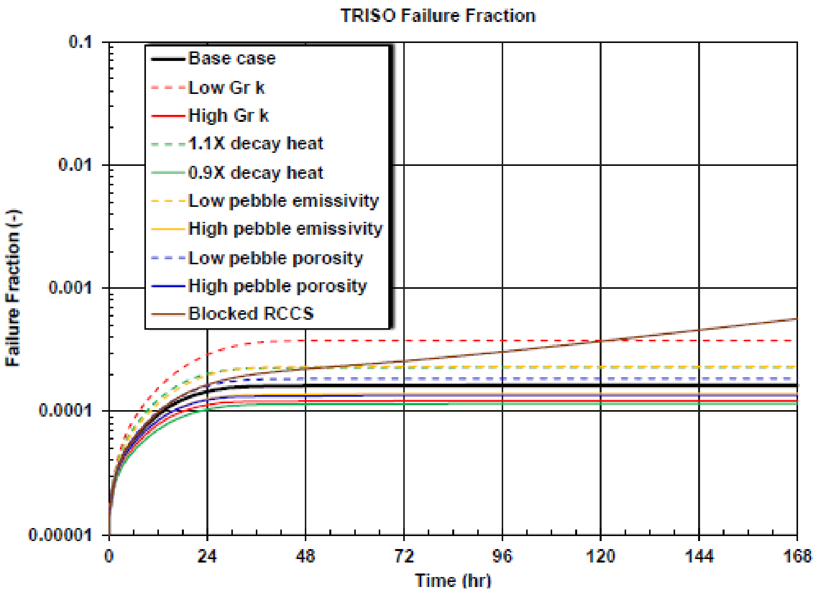
- Following scram, passive heat dissipation into reactor cavity ends the release from fuel
- Reactor building bypass requires two failures in a single heat pipe – one in the condenser region and another in the evaporator region
- Significant uncertainty in the release fractions depending upon the assumptions. No significance to the magnitude of release.

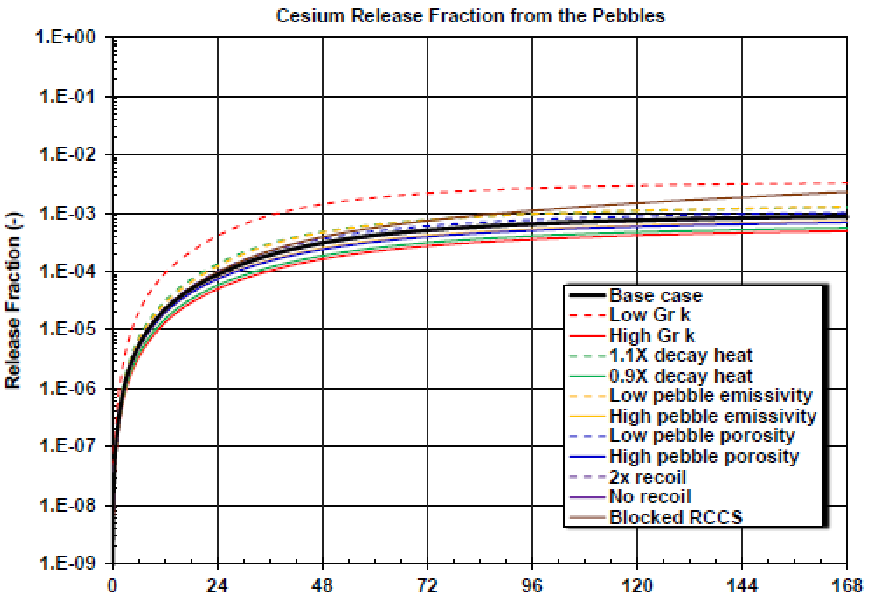


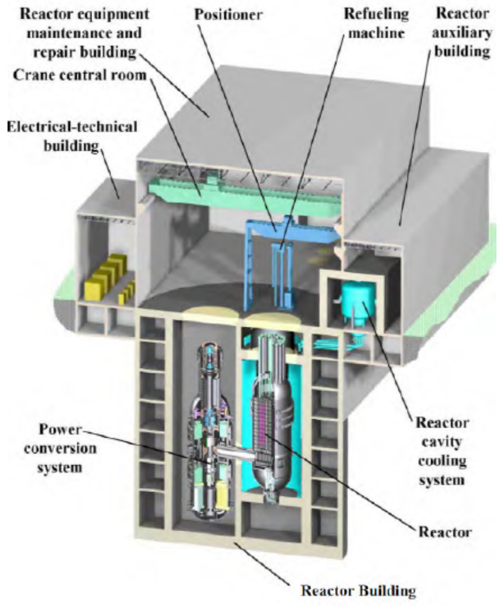




Pebble-Bed Gas-Cooled Reactor – PBMR-400







"HTGR Mechanistic Source Terms White Paper," July 2010, [INL-EXT-10-17999]

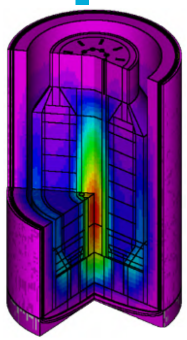
<u>Accidents Modeled</u>

- Depressurized loss-of-forced circulation

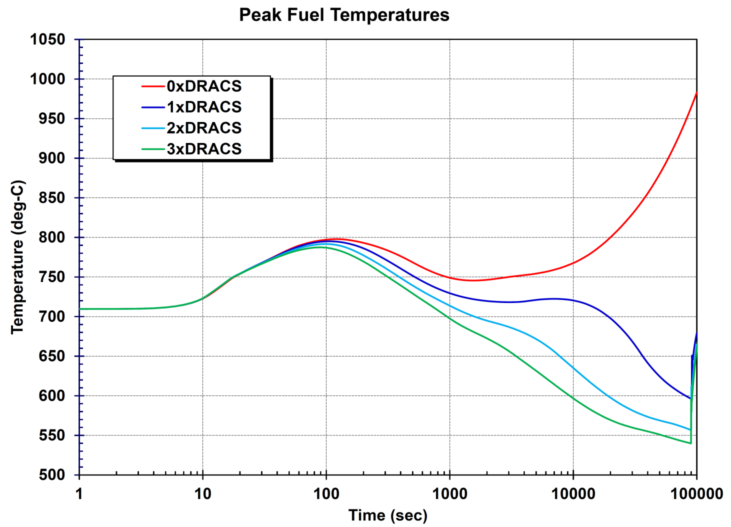
<u>Insights Gained on BDBA Behavior</u>

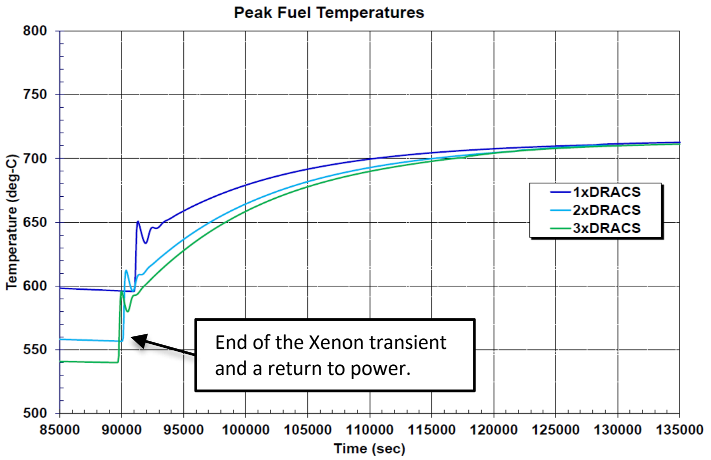
- Graphite oxidation from air ingress does not generate sufficient heat to impact fuel
- Passive heat dissipation into reactor cavity limits release from fuel failure
- A low graphite conductivity has the largest impact on the peak fuel temperature

Pebble-Bed Molten-Salt-Cooled – UCB Mark 1



ATWS with variable DRACS





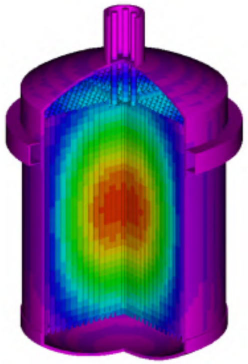
Accidents Modeled

- Anticipated transient w/o SCRAM
- Station-wide blackout
- Loss-of-coolant accident

Insights Gained on BDBA Behavior

- For ATWS, fuel heat-up was limited by reactivity feedback & passive decay heat removal system
- For SBO, with failure of the passive decay heat removal system, coolant boiling occurred over the course of several days
- For LOCA, with one train of decay heat removal system operating, coolant boiling was possibly averted.
- For LOCA, with failure of the passive decay heat removal system, fuel damage occurred.

Molten-Salt-Fueled Reactor – MSRE



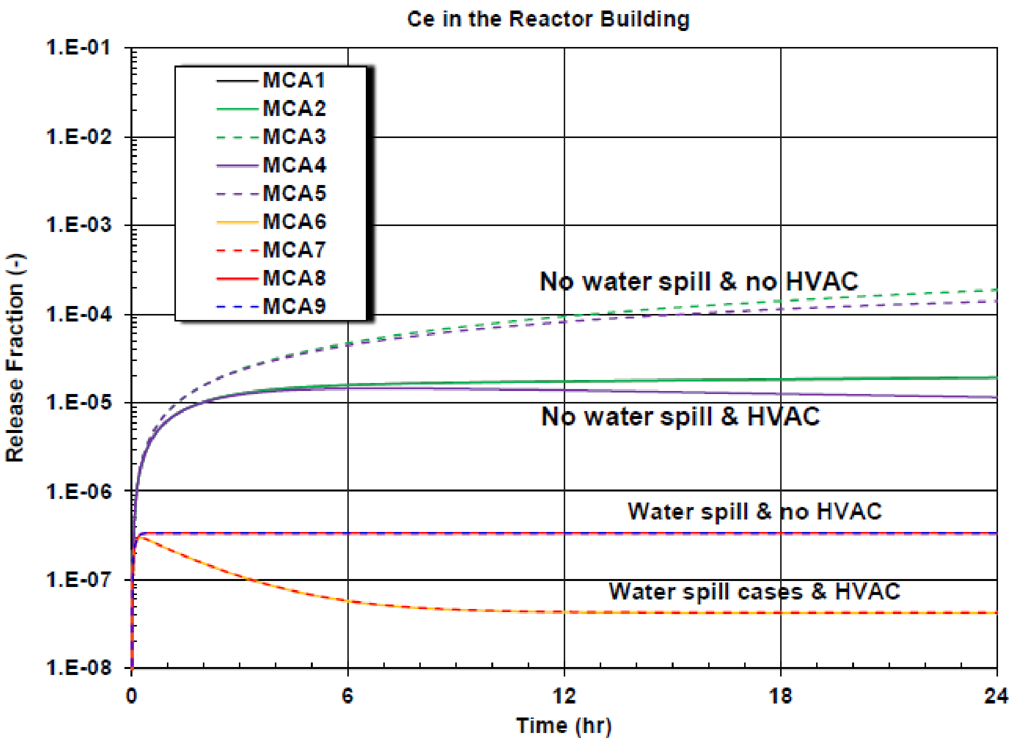
Accidents Modeled

- Full reactor inventory molten salt spill without water
- Full reactor inventory molten salt spill with water

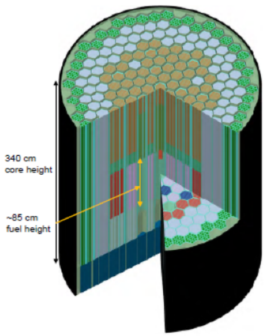
Insights Gained on BDBA Behavior

- Auxiliary filter operation increases the release rate of noble gases to the environment while also filtering airborne aerosols
- Aerosol releases to the environment are reduced due to settling in the reactor cell, capture in the filter, and capture in the condensing tank in the water spill cases
- The aerosol mass in the reactor building also spanned many orders of magnitude depending on scenario assumptions

All results of cerium in the reactor building



Sodium-Cooled Fast Reactor – ABTR

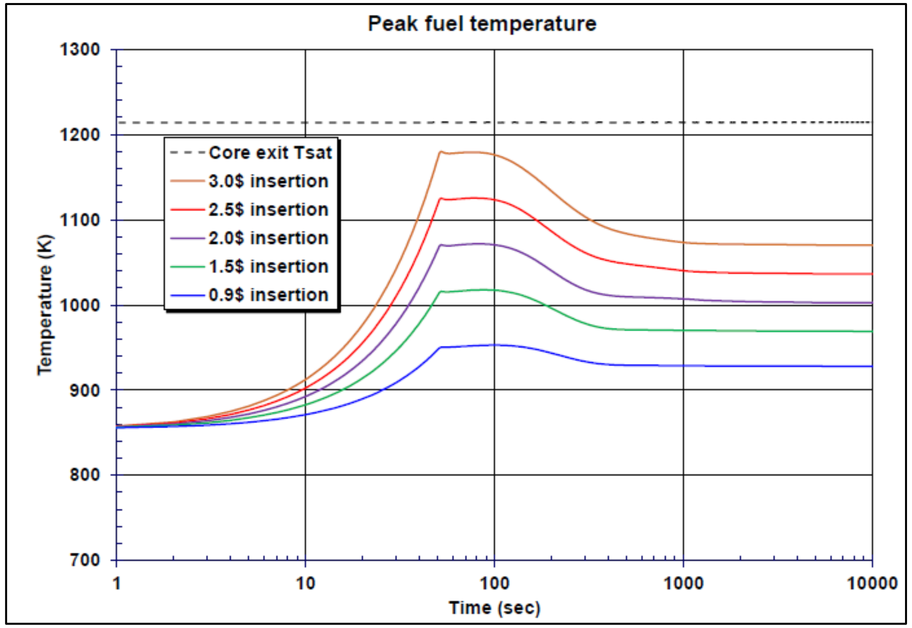


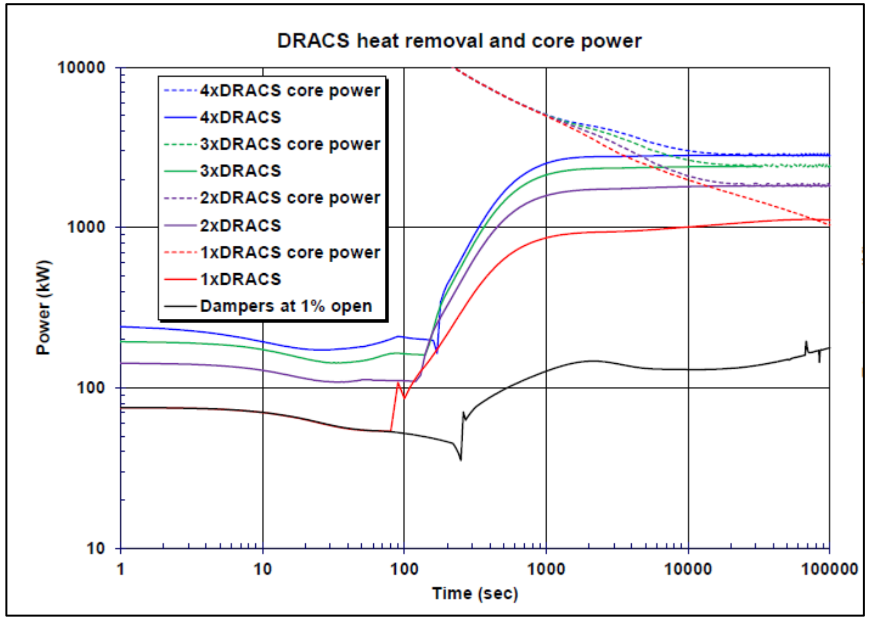
<u>Accidents Modeled</u>

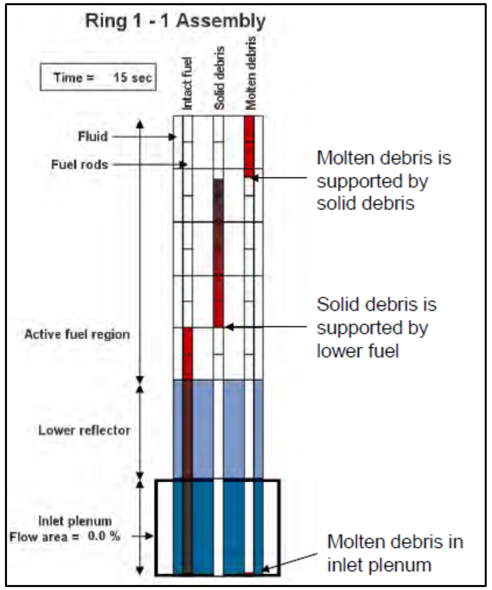
- Unprotected transient overpower
- Unprotected loss-of-flow
- Single blocked assembly

<u>Insights Gained on BDBA Behavior</u>

- With ULOF, core power eventually converges on the DRACS heat removal rate
- Single blocked assembly leads to rapid fuel damage







What have we learned & where are we going?

1. SCALE & MELCOR Volume 3 Models Support Readiness for NRC Non-LWR Licensing Reviews
   - Leveraged UCB Mk1 to Support NRR's review of the Hermes Construction Permit Application

2. Additional SCALE & MELCOR Code Enhancements & Capabilities In-Progress
   - Integration of SCALE's ORIGEN module into MELCOR for higher fidelity MSR transient analyses
   - Capability to model multiple working fluids in the same MELCOR plant model
   - Demonstrate capability for horizontal heat pipe reactors
   - Refinement of specialized models (e.g., fluid freezing and cascading heat pipe failures)
   - Fission product chemistry refinement

3. New Upcoming NRC Public Works for Additional Studies
   - SCALE & MELCOR Demonstration Calculations for a Molten Chloride Fast Spectrum Reactor
   - Public report – SCALE modeling of the sodium-cooled fast-spectrum ABTR
   - Public report – MELCOR Accident Progression and Source Term Demonstration Calculations for a Sodium-Cooled Fast-Spectrum Reactor

Volume 5

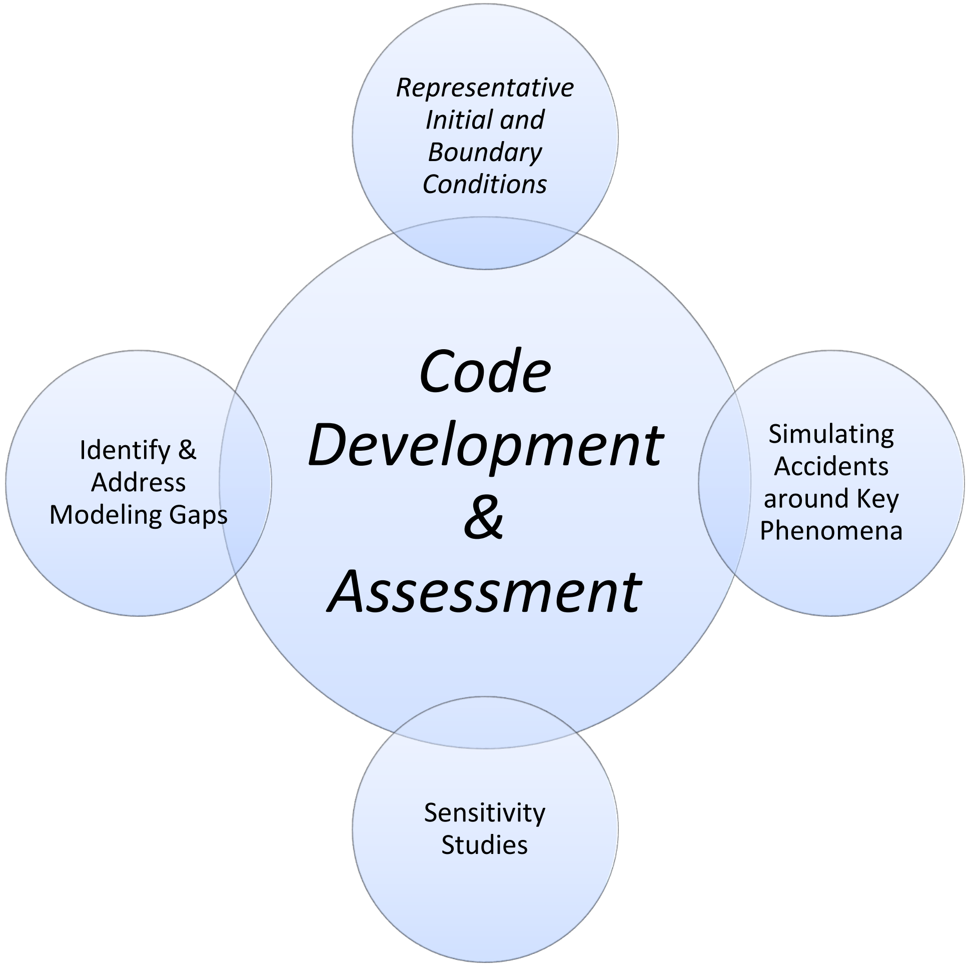
Radionuclide Characterization, Criticality, Shielding, and Transport for the Nuclear Fuel Cycle

Reminder: General Approach

1. Build SCALE core models and MELCOR models

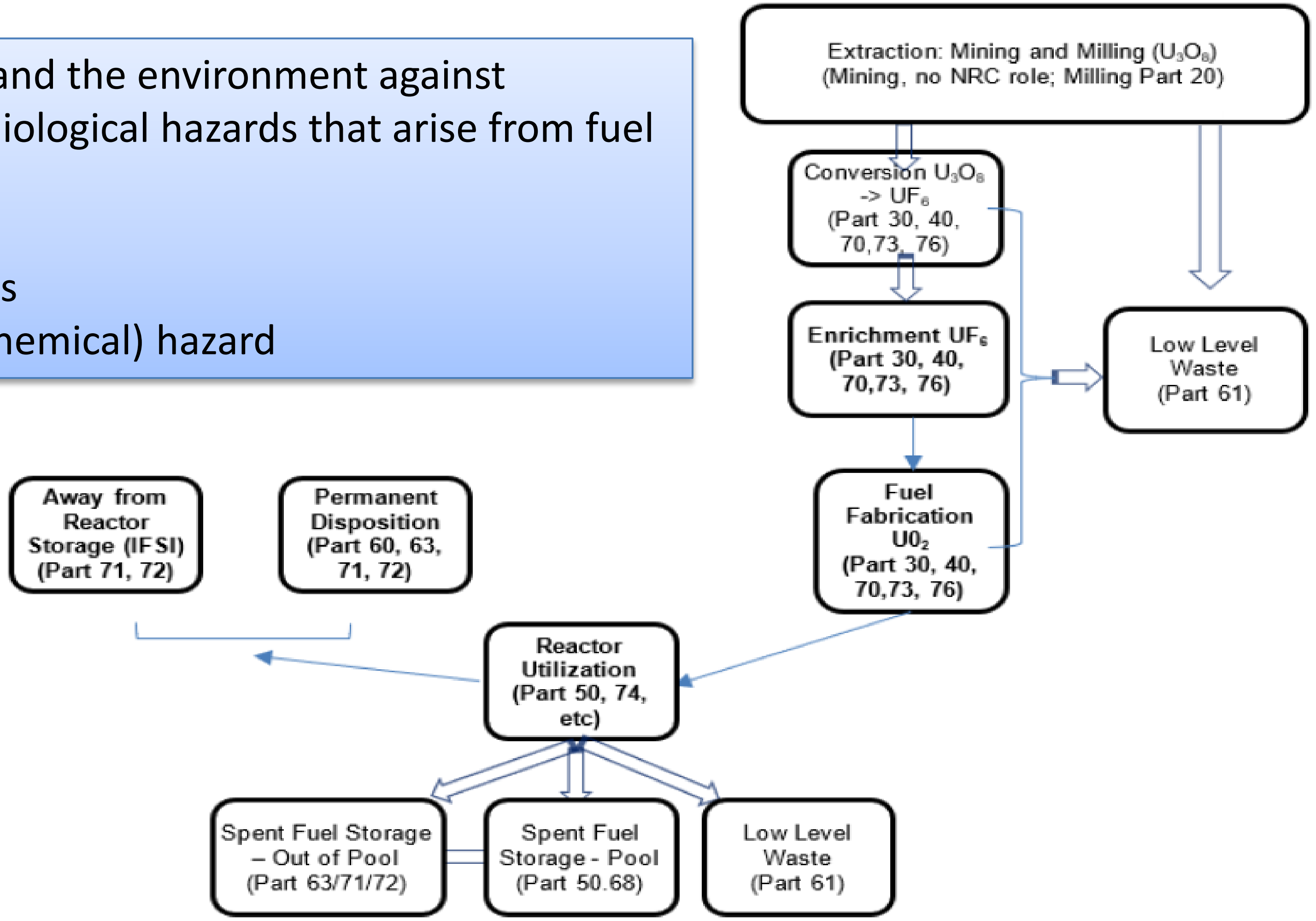
2. Select scenarios that demonstrate code capabilities for key phenomena

3. Perform simulations & code assessments on SCALE & MELCOR



LWR Nuclear Fuel Cycle & Regulations

- Protect workers, public and the environment against radiological and non-radiological hazards that arise from fuel cycle operations.
  - Radiation hazards
  - Radiological hazards
  - Non-radiological (chemical) hazard



Project Scope - Non-LWR Fuel Cycle

Stages in scope for Volume 5

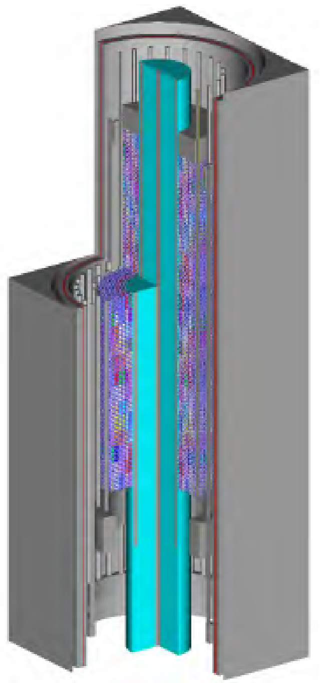
- Enrichment
  UF_6 enrichment
- UF_6 Transportation
- Fuel Fabrication
- Fresh Fuel Transportation
- Fuel Utilization (including on-site spent fuel storage)

Stages out of scope for Volume 5

Uranium Mining & Milling	• *Not envisioned to change from current methods.*
Power Production	• *Addressed in* ***<u>Volume 3 – Source Term& Consequence work</u>***
Spent Fuel Off-site Storage & Transportation	• *Large uncertainties & lack of information*
Spent Fuel Final Disposal	• *Large uncertainties & lack of information*

Non-LWR Designs Considered

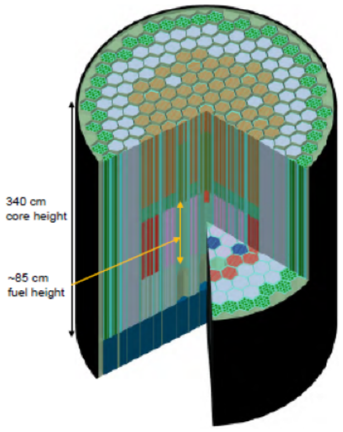
High-Temp. Gas Cooled Reactor



PBMR-400

- 400 MWth reactor, graphite moderated
- Helium-cooled & TRISO-particle pebble-fueled at 10 wt.% U-235
- Fuel discharged at high burnup (90 GWd/MTIHM)

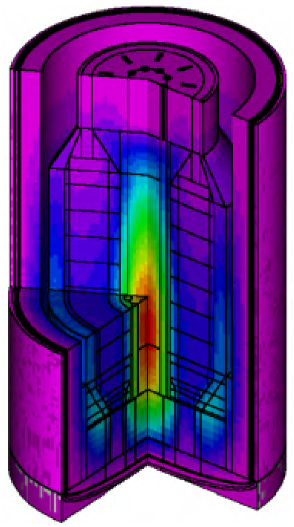
Sodium-Cooled Fast Reactor



ABTR

- 250 MWth pool-type reactor, utilizing metallic U / HT-9 fuel rods
- Reactor fueled with U-Pu-Zr fuel slugs
- Liquid sodium coolant

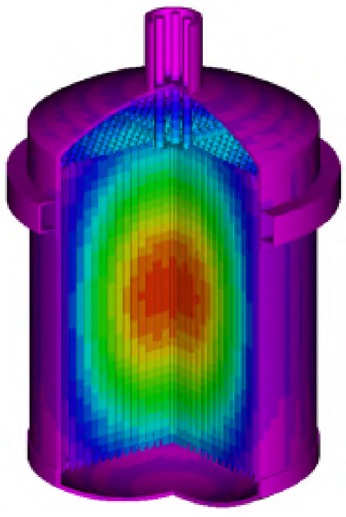
Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling

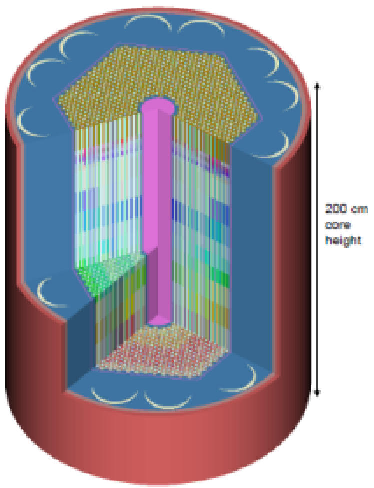
Molten Salt-Fueled Reactor



MSRE

- 10 MWth reactor, graphite moderated at near atmospheric pressures
- Reactor fueled with liquid dissolved fuel in molten salt (34.5 wt. % U-235)

Heat Pipe Reactor



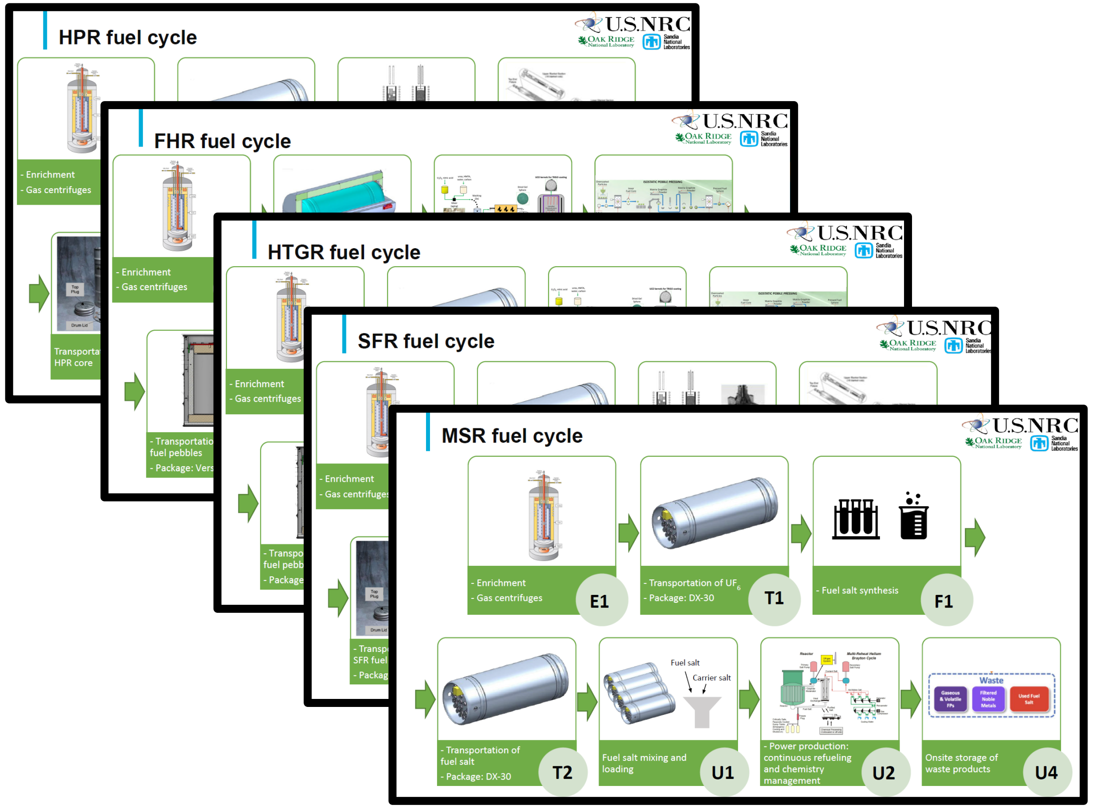
INL Design A

- 5 MWth with a 5-year operating lifetime
- 1,134 heat pipes fueled with UO_2 fuel (19.75 wt.% U-235)
- Reactivity controlled via control drums

Representative Fuel Cycle Designs

- Developed 5 non-LWR fuel cycle design concepts
  - HPR – INL Design A
  - HTGR – PBMR-400
  - FHR – UCB Mark 1
  - MSR – MSRE
  - SFR – ABTR

Designs to be documented in publicly available report later this year

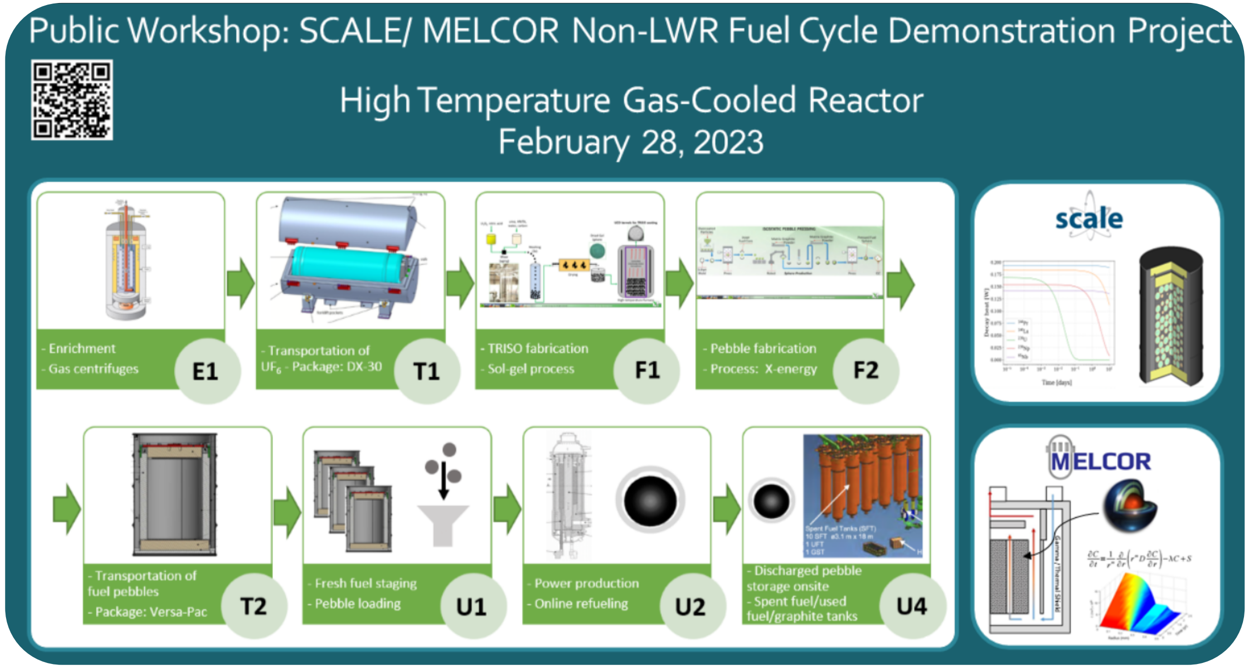


- Design concepts identify potential processes & methods
  - What shipping package could transport HALEU-enriched UF6? What are the hazards associated?
  - How is spent SFR fuel moved? What are the hazards associated?
  - How is fissile salt manufactured for MSRs? What are the various kinds of fissile salt that may be used? What are the hazards?

Used as the Initial and Boundary Conditions for developing SCALE & MELCOR models

non-LWR Fuel Cycle Demonstration Project –High Temperature Gas-Cooled Reactors

- *HTGR Fuel Cycle Highlights*
  - Use of HALEU (19.75 wt.% U-235)
  - No approved commercial-sized transport packages (UF6 & fresh pebbles)
  - New chemicals and processes for TRISO particle and pebble manufacturing
  - TRISO fabrication → Sol gel process ; pebble manufacturing
  - Continuous fuel circulation, loading, and removal

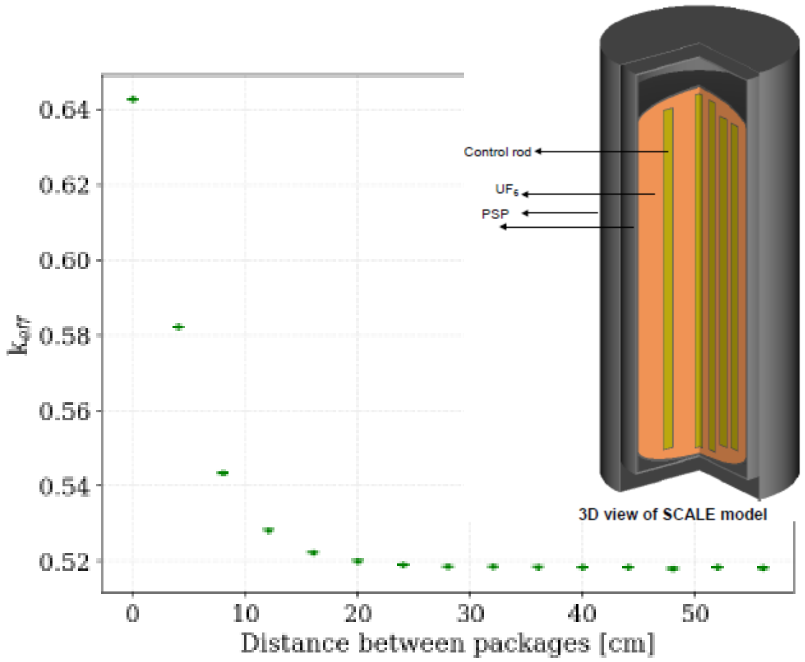


- *Accidents Modeled*
  1. Criticality due to HALEU-enrichments – UF_6 and fuel pebble operations
  2. Hazards associated with new chemicals (e.g., spills, water interaction, fire)
  3. Fission product release from damaged fuel pebbles during fuel handling

SCALE & MELCOR Analyses for Selected Accidents from the HTGR Fuel Cycle

Criticality-related Analyses

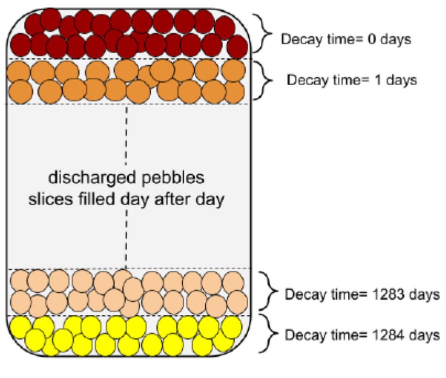
Water ingress into the DN30-X during UF_6 transport
- *Simulated UF_6 enriched to 10 & 20 wt. % U-235*
- *Shown to be subcritical*



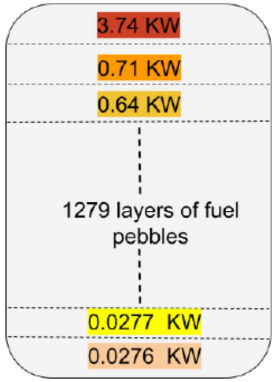
3D view of SCALE model

Spent Fuel Pebble Inventory & Fission Product Release

Spent Fuel Storage Tank Release
- *Spent fuel tank holding 620,000 pebbles simulated*
- *Approx. 500 pebbles discharged daily / 1284 days to fill SFT*
- *Total decay heat and inventory of SFT determined*

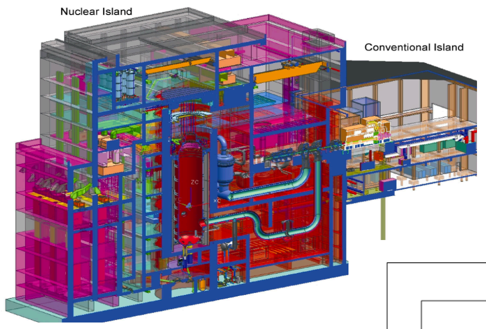


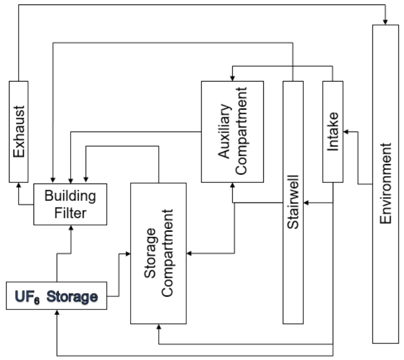
Spent fuel tank



In-Facility UF_6 Release

UF_6 Cylinder Rupture
- *UF6 cylinders are overfilled & heated → resulting in rupture and release*





Future Work for Volume 5

1. New Upcoming Planned Workshop
   - September 2023 – Sodium Fast Reactor Nuclear Fuel Cycle Analyses
   - 2024 – Molten Salt Fueled Reactor Nuclear Fuel Cycle Analyses

2. Upcoming Public Report(s)
   - Summer 2023 – Non-LWR Fuel Cycles for Severe Accident Simulations

3. Additional SCALE & MELCOR Code Enhancements & Capabilities In-Progress
   - New capabilities planned in SCALE for handling irregular geometries in SCALE (fuel reprocessing)
   - Leveraging newly developed capabilities to SCALE & MELCOR from Volume 3

Nuclear Supplier QA Program Qualification: ISO 9001 Supplemental Requirements

Mark Richter-Technical Advisor
Nuclear Energy Institute

July 20, 2023

NUCLEAR ENERGY INSTITUTE

Context

- Development plans for advanced reactors is of a scope and scale not before seen in our industry
- The current operating fleet must be supported for 40-60 years or more of safe and reliable operations
- The current supply chain will be challenged to meet the dynamic and growing demand as well as aggressive timelines for new parts and components
- Anticipated supply chain challenges will require new and transformative quality management approaches
- Opportunity for NRC to demonstrate regulatory leadership as a modern regulator, seeking new efficiencies in regulatory processes during a period of dynamic industry growth

ISO-9001 Approach to Meet Appendix B

- ISO-9001 is already employed in many other industries and some nuclear suppliers already have ISO-9001 programs

- NEI is developing a process whereby an ISO 9001 QA program, with enhancements, could be used as a framework for meeting the requirements of 10CFR50, Appendix B, leading to a more nimble and responsive supply chain. It is anticipated that this will be helpful in both maintaining the operating fleet as well as developing and deploying several hundred advanced reactors over the next decade.

- NOT proposing ISO 9001 as a replacement for 10 CFR Part 50 Appendix B

Not Starting From Scratch

- NRC SECY-03-0117 “Approaches for Adopting More Widely Accepted International Quality Standards” compares ISO-9001-2000 against the existing 10 CFR Part 50 Appendix B requirements and recommends that supplemental requirements would be needed
- EPRI 1007937 “Analysis and Comparison of ANSI/ISO/ASQ Q9001:2000 with 10CFR50 Appendix B: ISO-9001 Gap Analysis and EPRI 1002976, An In-Depth Review of Licensee Procurement Options for Use with ISO-9001 Suppliers”
- Other regulated industries utilizing an ISO-9001 based quality program and their regulating bodies have recognized the need for and implemented supplemental requirements

Implementation of NEI 22-04

- Decomposes each Appendix B criterion into discrete requirements and identifies comparable requirements from ISO 9001

- Note potential gaps for compliance with the regulation

- Provide recommendations for addressing the gaps and are implemented contractually by the Appendix B purchaser and potential ISO-9001 supplier

- Purchaser maintains 10 CFR Part 21 responsibilities, and supplier/vendor will impose the reporting requirements for nonconformances with sub-tier supplier(s)

Status and Next Steps

- NEI 22-04 “Nuclear Supplier QA Program Qualification: ISO 9001 supplemental requirements” rev. 0 is essentially complete

- ISO-9001 supplier dry run assessments informed final draft (Assessments of Pioneer Motor Bearing and Penn United Complete)

- NEI 22-04 undergoes broad industry review (Q3 2023)

- NEI participates in pre-submittal meeting with NRC (Q3 2023)

- NEI submits for NRC review and endorsement (Q4 2023)

Questions?

mar@nei.org

Lunch Break

Meeting will resume at 1:00 pm EST

[Microsoft Teams Meeting]

Bridgeline: 301-576-2978

Conference ID: 501 432 683#

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

NRC Staff Interactions with ACRS on Kairos Hermes Construction Permit Safety Review

Advanced Reactor Stakeholder Meeting
July 20, 2023

Matthew Hiser
Senior Project Manager
Advanced Reactor Licensing Branch 1
Division of Advanced Reactors and Non-Power Production and Utilization Facilities
U.S. Nuclear Regulatory Commission

Background

- Kairos submitted 11 topical reports prior to the Hermes construction permit (CP) application
  - All 11 topicals were approved prior to issuing the final Hermes CP safety evaluation
  - 8 of the 11 topical reports were reviewed by ACRS between 2020 and early 2023 (see Appendix II of ACRS letter)
- ACRS had strong familiarity with the Kairos technology and key technical topics involved in the Hermes CP application review

Timeline of ACRS Interactions on Hermes CP

- April 2022: Kairos and NRC staff presented Hermes CP overview
- January – March 2023
  - NRC staff provided preliminary Hermes CP SE chapters and key appendices to ACRS for review (all available in ADAMS under docket #05007513)
- March – May 2023: NRC staff and Kairos briefed ACRS on Hermes CP safety analysis and review
  - Kairos subcommittee: March 1, March 23-24, April 4, April 18
  - Full committee: May 3-4
- May 16, 2023: ACRS letter issued
- June 20, 2023: NRC staff response to ACRS letter issued

Staff Insights from ACRS Review of Hermes CP

- Staff appreciates the timely and thorough review of the Hermes CP application and safety evaluation

- ACRS used a risk-informed approach to focus on the most safety significant aspects of the design to ensure that the review was efficient and thorough

- ACRS review of preliminary safety evaluation chapters while the final safety evaluation was being assembled expedited ACRS review and accelerated project schedule

Questions?

Contact me by e-mail at Matthew.Hiser@nrc.gov or by telephone at (301) 415-2454

U.S. NRC ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) REVIEWS OF NEW FACILITY APPLICATIONS

Joy Rempe, Chairman

Advanced Reactor Stakeholder Meeting

July 20, 2023

ACRS Overview

- Provides Commissioners independent technical reviews of, and advice on, safety of proposed or existing reactor facilities, adequacy of proposed safety standards, and adequacy of NRC safety research program
  - Statutorily mandated by Atomic Energy Act of 1954, as amended
  - Operational practices governed by Federal Advisory Committee Act (FACA)
- Independent of NRC staff. Reports directly to the Commission, which appoints ACRS members

For additional information about ACRS, see: https://www.nrc.gov/about-nrc/regulatory/advisory/acrs.html

ACRS Review of Proposed Facility

- Integrated review of applicant submittals (and associated staff safety evaluation) including:
  - Safety Analysis Reports (for construction permit, operating license, early site permits, design certification, and standard design approvals)
  - Topical Reports and possibly other supporting documents (white papers, technical reports, etc.)
- Typically includes one or more subcommittee meetings and at least one full committee meeting prior to issuance of ACRS letter report.
  - Portion of meetings open to public, allowing opportunities for public comments
  - Portion of meetings may be closed to allow discussion of proprietary information

ACRS Review of Proposed Facility (cont'd)

- ACRS developing 'best-practices' guidance* to promote streamlined reviews focused on safety and risk significant aspects
  - Implements lessons learned from recent design-centered reviews
  - Lead ACRS member and ACRS staff work with cognizant NRR staff to develop committee engagement plan to optimize review schedule
  - ACRS review completed *after* staff draft safety evaluation report completed**
- Typically includes topics*** such as: overall design (emphasizing unique and novel aspects), safety functions and principal design criteria, safety-related structures, systems, and components, licensing basis event selection, fuel qualification, safety analysis methods and results, and source term.

*Guidance, along with name of lead ACRS member for each design-centered subcommittee to be posted on https://www.nrc.gov/about-nrc/regulatory/advisory/acrs.html.

**ACRS reviews may be concurrent with staff completion of some safety evaluation report chapters.

***Representative topics, not comprehensive.

Advanced Reactor Population-Related Siting Considerations

July 20, 2023

NUCLEAR ENERGY INSTITUTE

Acronyms

- BWR – Boiling Water Reactor
- CFR – Code of Federal Register
- DiD – defense-in-depth
- EAB – exclusion area boundary
- LPZ – low population zone
- LWR – light water reactor
- MWth – megawatt thermal
- PCD – population center distance
- PDD – population density distance
- ppsm – persons per square mile
- PWR – Pressurized Water Reactor
- SAMA – Severe Accident Mitigation Alternatives
- SECY – Commission paper
- SOARCA – State-of-The-Art Reactor Consequence Analysis
- SRM – Staff Requirements Memorandum

Background

- 10 CFR Part 100.21 requirement:

  “(h) Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable[3].”

Guidance for Implementation

- Regulatory Guide (RG) 4.7, “General Site Suitability Criteria for Nuclear Power Stations”
  - Current criterion: population density not exceeding 500 persons per square mile (ppsm) out to 20 miles
  - Based on large light water reactor experience
- SRM-SECY-20-0045 directed NRC staff to revise RG 4.7 guidance relating to 10 CFR 100.21(h) to include provisions for advanced reactor designs
  - No greater than 500 ppsm out to twice the distance at which 1 rem dose for the 30-day exposure period is calculated based on design-specific events

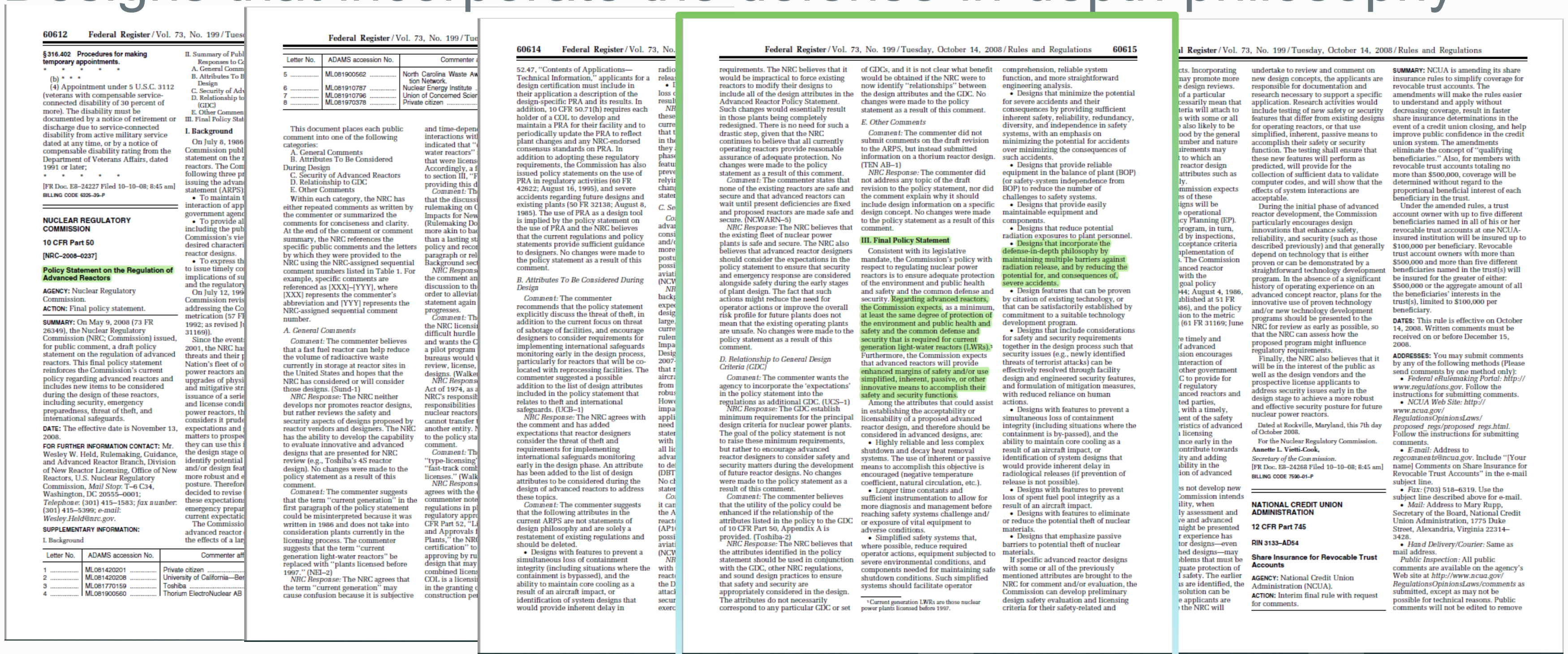
Industry's Effort to Provide Feedback

- Goal – Assess NRC's pre-decisional white paper, "Alternative Approaches to Address Population-Related Siting Considerations" made public in April 2023 and provide industry's observations to inform further discussions
- Objectives
  - Put the population-density siting consideration in context with other siting elements and defense-in-depth considerations
  - Compare the level of protection afforded as proposed by NRC for advanced reactors to that currently applied to existing LWRs
  - Identify whether NRC's guidance would result in undue burden (i.e., excessive restrictions on siting) for advanced reactors
- Scope – Cover all advanced reactor designs except the following
  - Large (gigawatt scale) designs

Population-Density Siting Consideration in Context

NRC Advanced Reactor Policy Statement

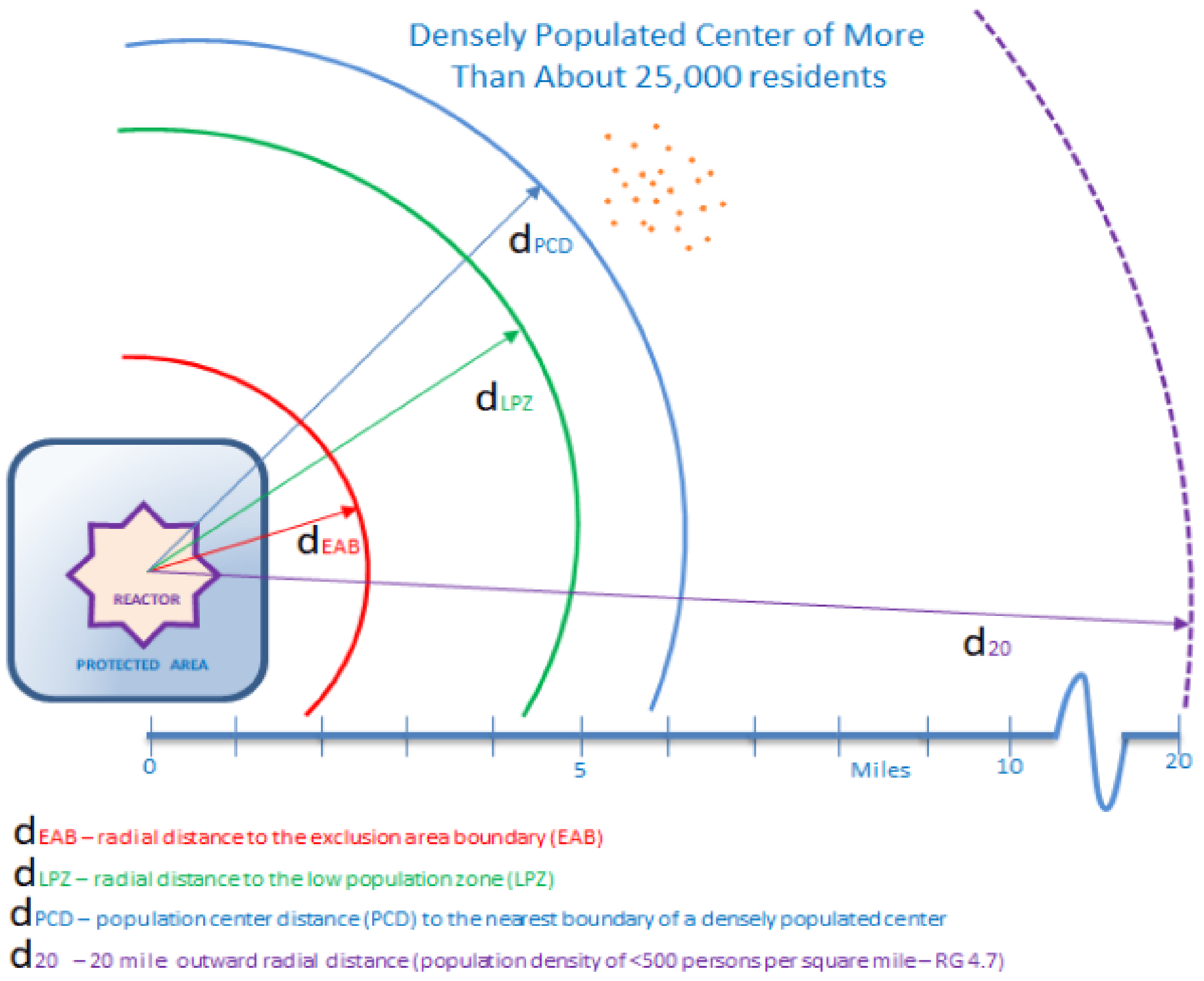
- At least the same degree of protection of the environment, public health & safety, and common defense and security
- Enhanced margins of safety and/or use of simplified, inherent, passive or other innovative means to accomplish safety & security functions
- Designs that incorporate the defense-in-depth philosophy



Siting Criteria / Limitations

- Power reactor siting has typically involved assessment of a variety of distances, most of which are depicted in Fig. 1
- Each provides functional and defense-in-depth (DiD) purposes
- Siting criteria protect from societal impacts & provides DiD to minimize societal impacts should containment fail*

*as we understand it; based on TID 14844 (1962)



Source: NRC SECY 20-0045 Figure 1

Siting Criteria / Limitations (con't)

- Prior to 1973, PDD was not considered (no $d_{"20"}$ – only d_{EAB} , d_{LPZ} and d_{PCD})
- WASH-1308 (1973) suggested need for an RG and that there should be a PDD, but different approach (equivalent to ~1600 ppsm)
- RG 4.7, R0 (1974) did not include PDD (no $d_{"20"}$ – only d_{EAB} , d_{LPZ} and d_{PCD})
- RG 4.7, R1 (1975) added PDD, but it was d_{30}
  - 500 ppsm, different than what was proposed in WASH-1308
  - No reference to a dose basis
- RG 4.7, R2 (1998) changed PDD to d_{20}
  - No reference to a dose basis
  - States only that "Numerical values in this guide are generally consistent with past NRC practice and reflect consideration of severe accidents, as well as demographic and geographic conditions characteristics of the United States."
- ORNL/TM-2019/1197 (2019) evaluated various alternatives to the current requirements that would achieve a reduction of about an order of magnitude versus large LWRs, including ratio of thermal output and case-be-case, design-specific review, but did not recommend a specific dose criterion

Compare the Level of Protection

How does the proposed PDD criterion compare to the existing 20-mile PDD for Large LWRs?

- Existing basis for 20-mile PDD
  - Deterministically set for current large LWRs; no calculation of the dose at that distance required.
  - NRC states in the pre-decisional white paper that the 20-mile distance was based on insights from probabilistic risk assessments and other studies associated with light-water reactor designs.
  - Have not found documentation for original basis other than as summarized on previous slide.
- NRC proposed alternative advanced reactor PDD
  - Distance that is “equal to twice the distance at which a hypothetical individual could receive a calculated TEDE of 1 rem over a period of 1 month from the release of radionuclides following postulated accidents”
  - We could not find a clear technical basis for this criterion and so the basis is not well-understood
  - Dose at PDD (twice the distance to 1 rem) will be much less than 1 rem, on the order of doses from background radiation

What model inputs and assumptions does NRC expect applicants to use?

- Some do not appear to be specified in NRC's pre-decisional white paper, "Alternative Approaches to Address Population-Related Siting Considerations"
- Based on past practice, we assumed the NRC might expect the most conservative case
  - No credit for shielding with normal activity; individual is modeled as residing outdoors unprotected for 30 days
  - 95% Weather – WD Maximum; very conservative weather assumptions
- Note: industry does not believe the worst case is necessary, but wanted to start from the worst case that we believed the NRC might expect
  - In any case, we compare models done with consistent assumptions

Population Density Distance (PDD) Dose Historical Effectiveness

- Jensen Hughes conducted a scoping atmospheric dispersion consequence analysis using the WinMACCS code
- Scoping model purpose: to estimate dose calculated using the existing (LWR based) PDD siting criterion (i.e., the level of protection actually provided by the 20-mile limit for the current fleet) for comparison with the NRC proposal for advanced reactors of twice the distance to 1 rem dose over 30 days

LWR Scoping WinMACCS Model Inputs and Assumptions

- Started with NRC Linear No-Threshold (LNT) Point Estimate Sample Problem distributed with WinMACCS version 3.10
- Core inventory taken from the NRC's SOARCA for the Surry reactor and ratioed up by 50% (effectively representing 3819 MWth)
  - reasonably representative of a "typical" large LWR
- Radionuclide release magnitudes (to the environment) based on calculating an average of the total release fraction of the highest releases from 13 recent SAMA analyses (both PWR and BWR) using a frequency screening
  - WinMACCS typical three plume model/assumptions used for total release
- Surry meteorology data per example data with WinMACCS code
- Dose was calculated for 30 days, with no credit for protective actions

LWR Scoping WinMACCS Model

- Eight cases
  - Cases 1 – 4 represent the 30-day dose for individuals who maintain normal activity during the release and thereafter
  - Cases 5 – 8 eliminate credit for shielding with normal activity; individual is modeled as residing outdoors unprotected for 30 days
  - Cases reflect four different levels of weather conditions assumptions
- Case 8 results are the most restrictive

LWR Scoping WinMACCS Model Results

- Observation: The Commission has previously stated that the current generation of plants is adequately safe
  - Current 20-mile distance results in acceptable societal risks
  - Doses for large LWR at 20 miles for 30 days do not pose undue health risks
- Applying NRC proposed twice the distance to 1 rem over 30 days
  - This dose criterion is orders of magnitude less than what is currently accepted by NRC
  - Distances produced using this criterion would be an order of magnitude more than 20 miles
  - NRC proposed criterion is excessively restrictive on siting

Advanced Reactor Scoping WinMACCS Model

- Estimates calculated for two advanced reactor designs based on the generic LWR WinMACCS model (assumed representative of others)
  - High-Temperature Gas Reactor with tri-structural isotropic (TRISO) fuel
  - Molten Salt Reactor
- Release-related inputs for these scoping HTGR and MSR models were based on data developed by Sandia in SAND2020-0402 for performing simplified scoping assessments of advanced reactors
- Used a postulated maximum credible accident (MCA) based upon review of the Sandia study
  - Small core inventory, e.g., 250 MWth
  - Assumed percentage of fuel damage; percentage of fission product migration from fuel to environment; degraded containment
- Dose was calculated for 30 days, with no credit for protective actions (same as large LWR model)

Advanced Reactor Scoping WinMACCS Model Results

- Observations from large LWR model results (slide 16) also applicable to advanced reactors
- NRC proposed PDD estimates for advanced reactors would be excessively more conservative than what the NRC currently finds acceptable for large LWRs
- If the current level of protection for the existing LWR fleet is used, the PDD for these advanced reactors could be well within a site boundary
- NRC's proposed PDD approach would create excessive restrictions on the ability to site advanced reactors – far beyond what NRC imposes on large LWRs today when the comparative size of source terms in considered

Identify Whether NRC Approach Results in Undue Restrictions on Siting

Does NRC approach Appropriately Credit Safety Features of Advanced Reactors?

- SRM-SECY-20-0045 approved the staff's recommended option to revise the guidance in RG 4.7 to include technology-inclusive, risk-informed, and performance-based criteria for population densities that are based on estimates of radiological consequences from design-specific events
  - NRC's proposed PDD criterion would result in siting restrictions that do not reflect the potential for enhanced safety and reduced risks associated with radiological releases from advanced reactor designs
  - More guidance needed to balance realistic/conservative assumptions
  - The impact on micro-reactors was not analyzed, but can be anticipated as consistent with other observations
  - Layers of DiD applied in the design deserve credit, as applicable, e.g., small source term, slow accident progression, low operating pressure

Summary Observations

Key Take-Aways for Further Discussion

- Alternate dose criterion should be developed to be more representative of the currently accepted level of protection for large LWR licensing.
    - The proposed criterion associated with very low frequency events is excessively conservative compared to previous large LWR licensing and compared to the annual exposure of the public from natural and man-made sources (e.g., medical procedures).
    - The undue burden created by excessive conservatism significantly restricts advanced reactor siting as compared to the NRC's currently accepted approach for large LWRs.
- Clarity is needed on the modeling assumptions, which heavily influence dose criterion calculations, including consideration with respect to the realistic exposure risk to the public that would be acceptable to the NRC
- NEI is preparing a white paper with more detail on our observations to inform NRC's consideration of revisions to draft guidance.

Break

Meeting will resume at 2:45 pm EST

[Microsoft Teams Meeting]

Bridgeline: 301-576-2978

Conference ID: 501 432 683#

MACCS Consequence Analysis Demonstration Calculations

Advanced Reactor Stakeholders Meeting
July 2023

AJ Nosek, PhD
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U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Purpose

- The purpose of this project was to:
  - Assess the capabilities of the MACCS code to analyze a selected conceptual advanced reactor design under a postulated accident scenario.
  - Identify potential gaps that may exist in conducting such an analysis, both technical and practical.
  - Exercise new models (e.g., nearfield models) and new settings (i.e., inventory and source term) in the MACCS code for the selected advanced reactor design.

- The project continued similar code readiness work using SCALE and MELCOR:
  - Oak Ridge National Lab (ORNL) developed a SCALE model to compute the radionuclide core inventory of the Idaho National Lab (INL) Design A conceptual reactor design.
  - Sandia National Labs (SNL) developed a MELCOR model to simulate postulated accidents of the INL Design A conceptual reactor design.

Project Approach

- All MACCS analyses used the following inputs:
  - An example inventory from a demonstration SCALE model (ORNL/TM 2021/2021)
  - An example source term and reactor building dimensions from a MELCOR demonstration model (SAND2022-2745)
  - Weather and regional characteristics from an existing (Sequoyah Nuclear Plant) site
  - Other general settings defined in the MACCS parameter guidance report (NUREG/CR-7270)

Project Approach

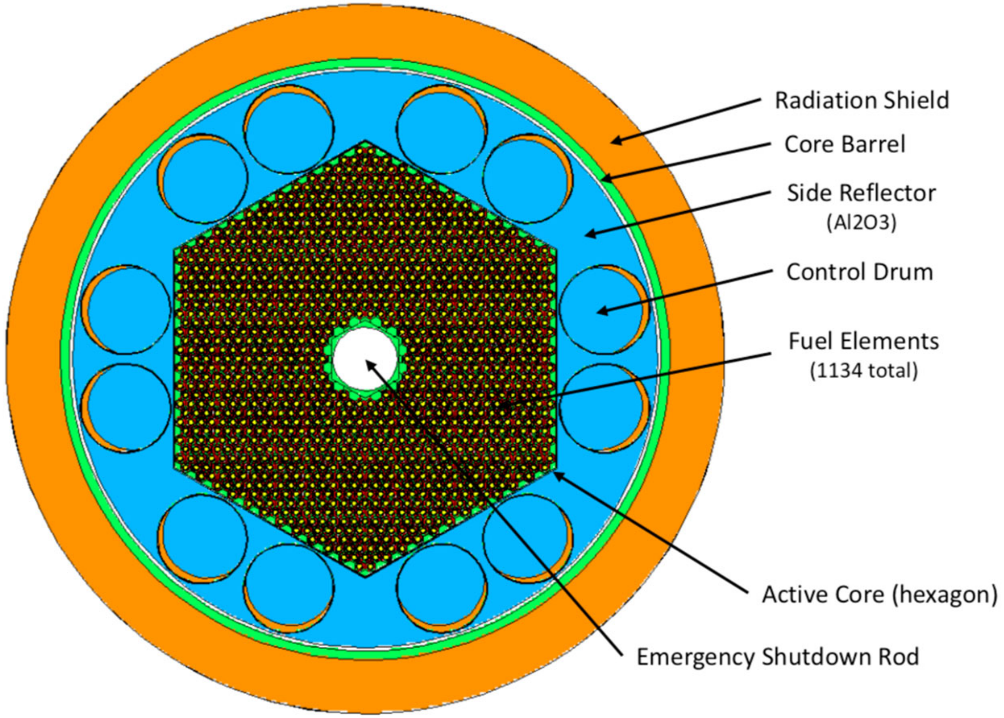
- The main analysis used the following settings:
  - A radionuclide list for consequence analysis from SAND2022-12018 based on example inventory from the demonstration SCALE model
  - The Regulatory Guide 1.145 full model for nearfield transport
- Sensitivity analyses evaluated the following characteristics:
  - Radionuclide lists for consequence analysis
  - Dose exposure periods
  - Nearfield models
  - Release timings

Project Approach

- The project used MACCS v4.1.
- All reported doses are projected doses in the ambient environment.
- The MACCS calculations sample from a range of weather conditions as input. The mean, 5^{th} quantile, and 95^{th} quantile MACCS outputs represent the distribution of results due to weather uncertainty.

Example Accident Scenario Description

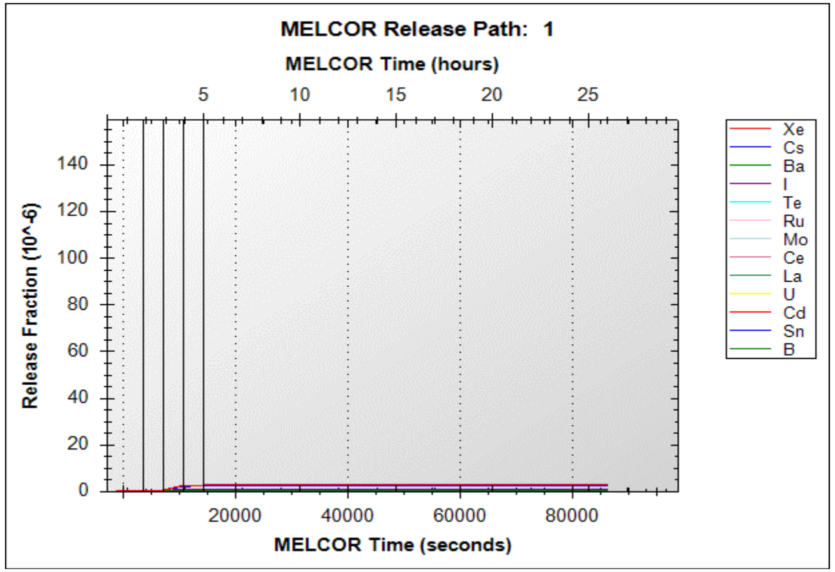
- To demonstrate code capabilities, the MELCOR team selected accident conditions for the MELCOR INL Design A model.
- This effort produced an example accident progression and source term for a transient overpower (TOP) accident scenario.
- The MACCS calculations use the example atmospheric release from the MELCOR demonstration project as input
- The MACCS demonstration calculations do not reflect realistic radiological consequences outside of the conditions assumed in the MELCOR analysis.

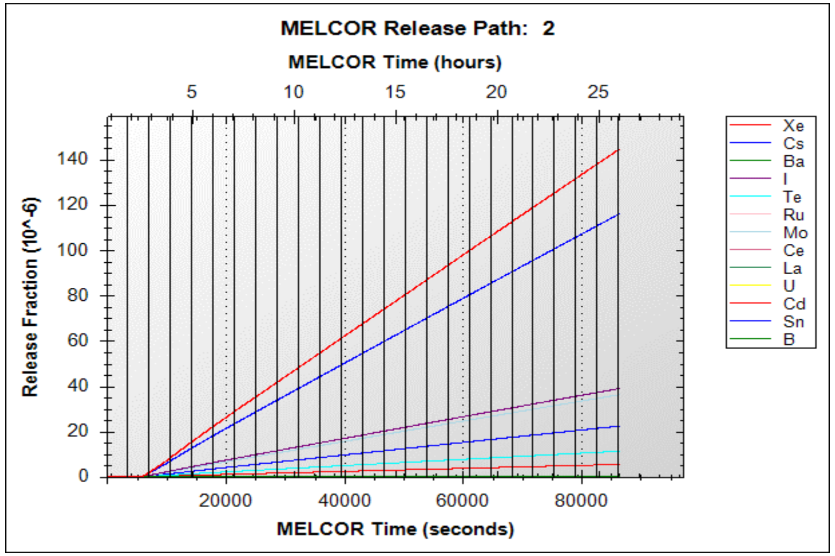


INL Design A reactor vessel cross-section (from figure 3-1 of SAND2022-2745)

Source Term*

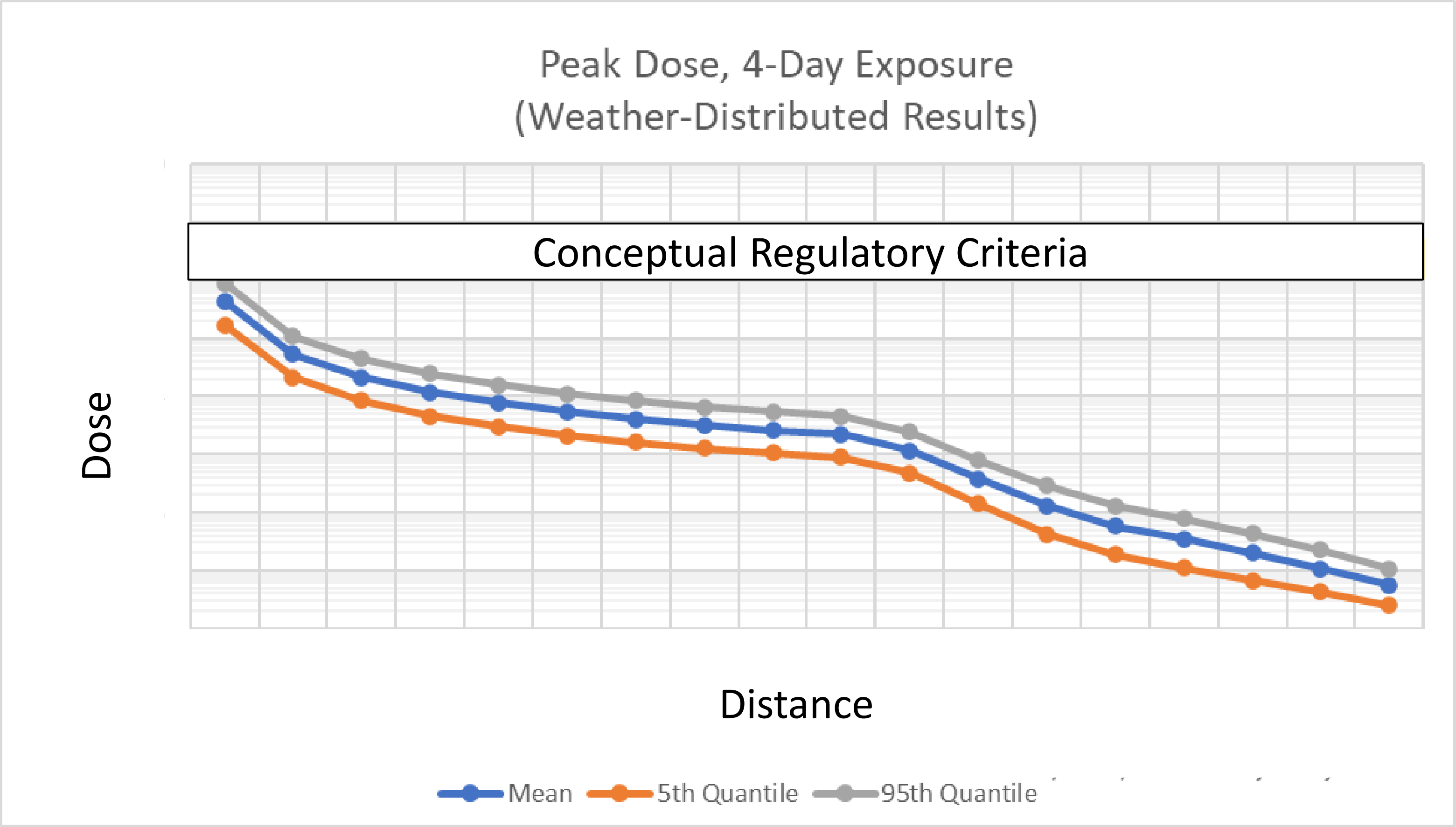
- Two release pathways from building at 5.15 m and 9.15 m (16.9 ft and 64 ft).
- Release begins in less than 1 hour.
- Relatively small amount of release before MELCOR cutoff time of 24 hours.
- Project divided plume into 28, 1-hour segments.





*Note: Consequence results are shown to illustrate code capabilities only. Actual consequences would be based on design, site, and scenario-specific factors.

Example of Main Analysis Results*



*Note: Consequence results are shown to illustrate code capabilities only. Actual consequences would be based on design, site, and scenario-specific factors.

Sensitivity Analysis of Radionuclides

- MACCS guidance recommends using a subset of radionuclides from the LWR core for analysis
- That list was expanded to include additional radionuclides from the conceptual HPR core

Recommended Radionuclide List for LWR Applications (from NUREG/CR-7270, Table 2-2)

Co-58	Y-90	Ru-103	Te-132	Ba-137m	Nd-147
Co-60	Y-91m	Ru-105	I-131	Ba-139	Np-239
Kr-85	Y-91	Ru-106	I-132	Ba-140	Pu-238
Kr-85m	Y-92	Rh-103m	I-133	La-140	Pu-239
Kr-87	Y-93	Rh-105	I-134	La-141	Pu-240
Kr-88	Zr-95	Rh-106	I-135	La-142	Pu-241
Rb-86	Zr-97	Te-127	Xe-133	Ce-141	Am-241
Rb-88	Nb-95	Te-127m	Xe-135	Ce-143	Cm-242
Sr-89	Nb-97	Te-129	Xe-135m	Ce-144	Cm-244
Sr-90	Nb-97m	Te-129m	Cs-134	Pr-143	Sb-127
Sr-91	Mo-99	Te-131	Cs-136	Pr-144	Sb-129
Sr-92	Tc-99m	Te-131m	Cs-137	Pr-144m	

Additional Radionuclides from draft SAND2022-12018, Table 4-2

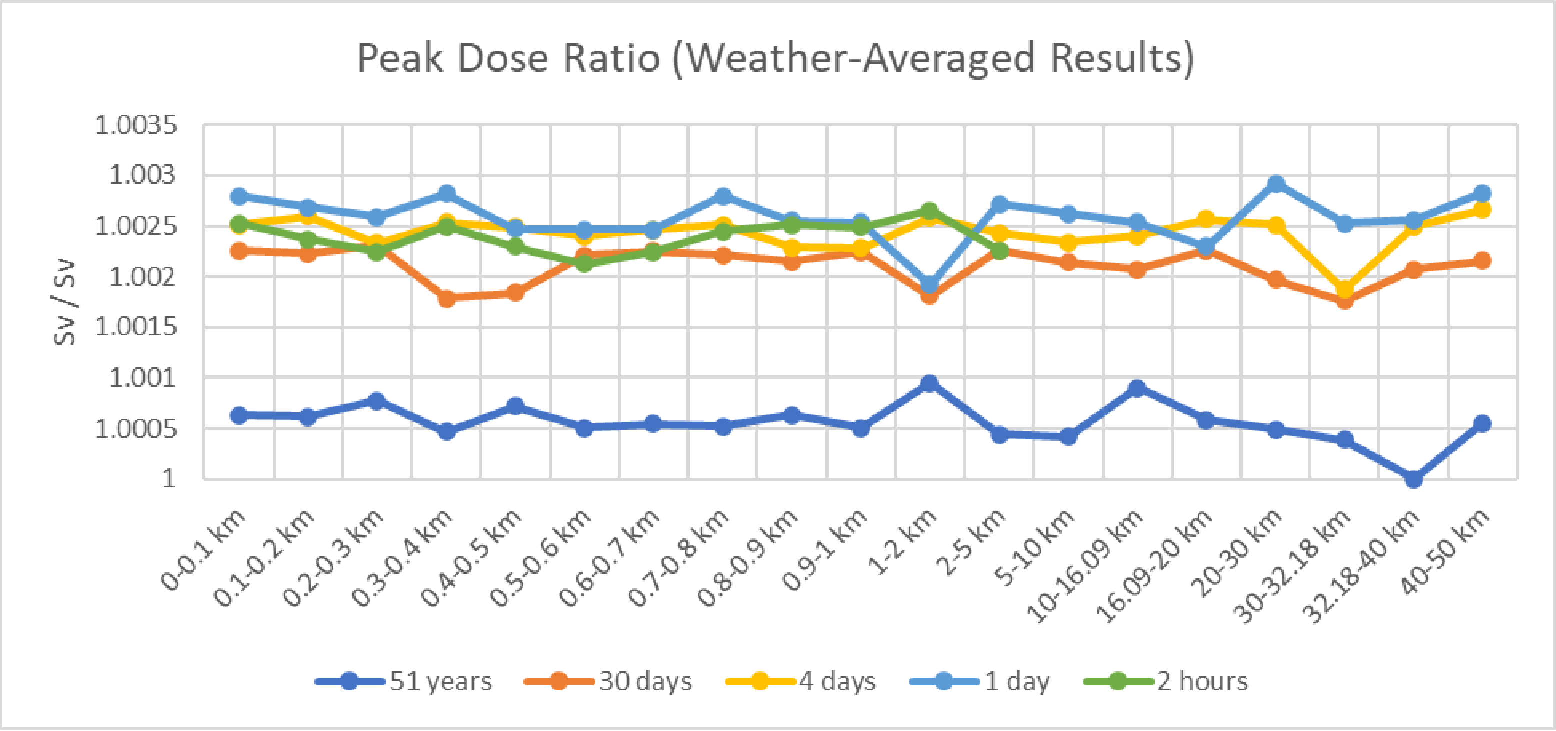
Ag-111	Ge-77	Pm-149	Sn-121
As-77	Nb-95m	Pm-151	Sn-123
Cd-115	Nd-149	Pr-145	Sn-125
Cd-115m	Pd-109	Sb-125	Sn-127
Eu-154	Pm-147	Sm-151	Te-125m
Eu-155	Pm-148	Sm-153	U-234
Eu-156	Pm-148m	Sm-156	U-237

Additional Daughter Radionuclides

In-115m	In-115	Th-230	Ra-226	Rn-222

Sensitivity Analysis of Radionuclides*

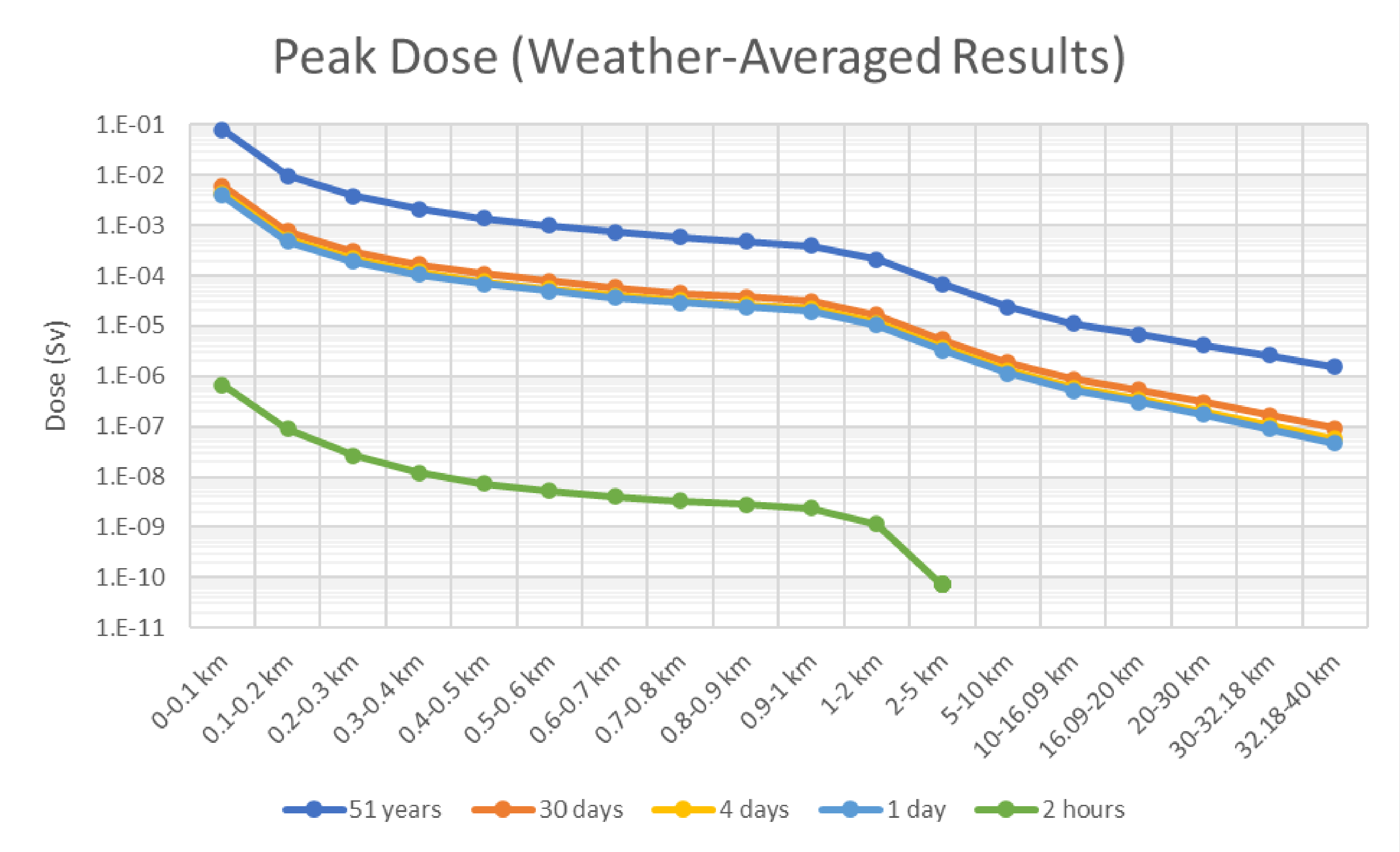
- A sensitivity using the updated radionuclide list shows little change in the results.
- The ratio of the peak doses from the new radionuclide list compared to the LWR list shows minimal increase in consequence.
- Note: this did not evaluate radionuclides important to ingestion consequences, which is based on a separate list



*Note: Consequence results are shown to illustrate code capabilities only. Actual consequences would be based on design, site, and scenario-specific factors.

Sensitivity Analysis of Dose Exposure Period*

- The analyzed exposure periods begin at the start of the accident. Release begins quickly at less than 1 hour.
- Longer exposure periods cause greater doses.
- At 2 hours, only a fraction of the release has occurred, and no plume segments have travelled beyond 5 km (3.1 mi).



*Note: Consequence results are shown to illustrate code capabilities only. Actual consequences would be based on design, site, and scenario-specific factors.

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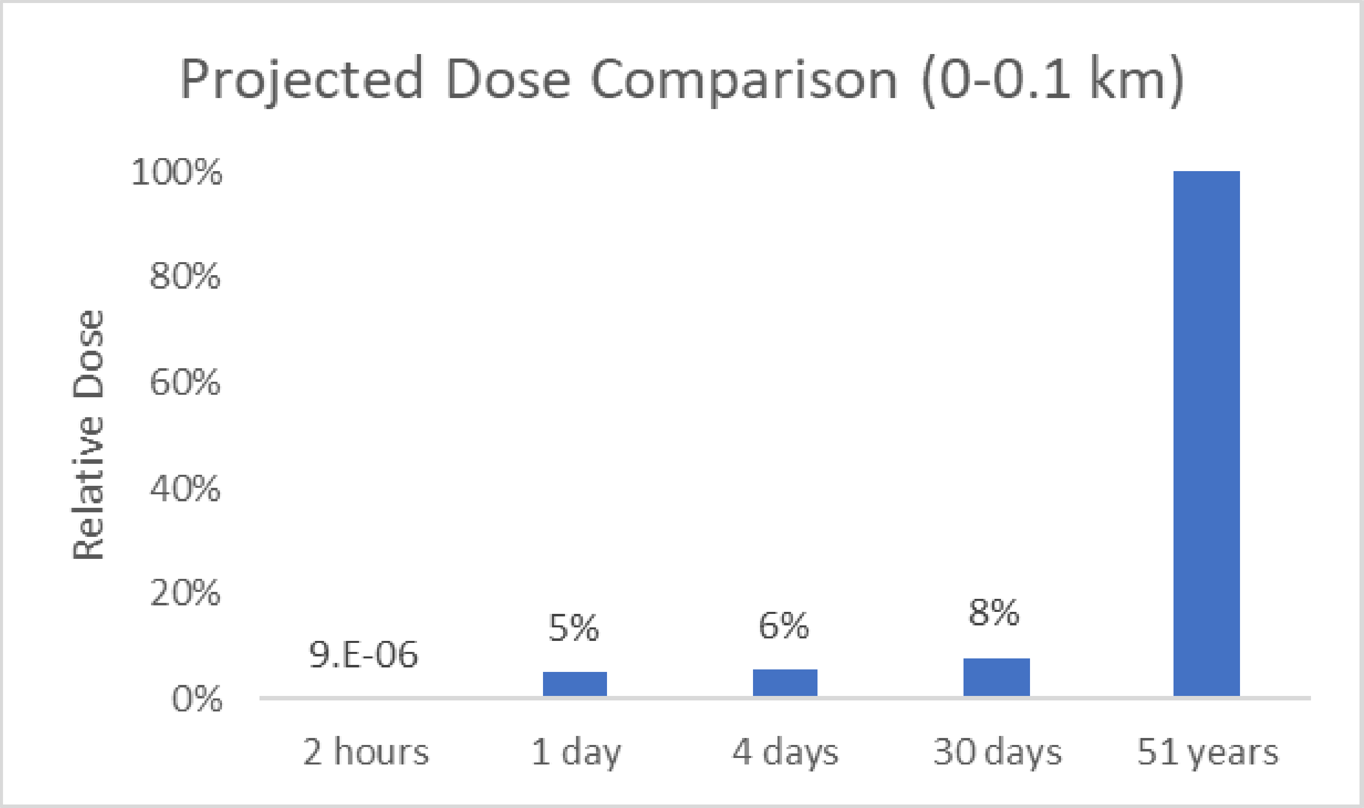
Sensitivity Analysis of Dose Exposure Period*

- Exposures during plume passage and exposures to short lived radionuclides occur only in the early phase of the accident.
- Nevertheless, exposure to ground contamination over the long term is the dominant contributor to the overall lifetime dose projection



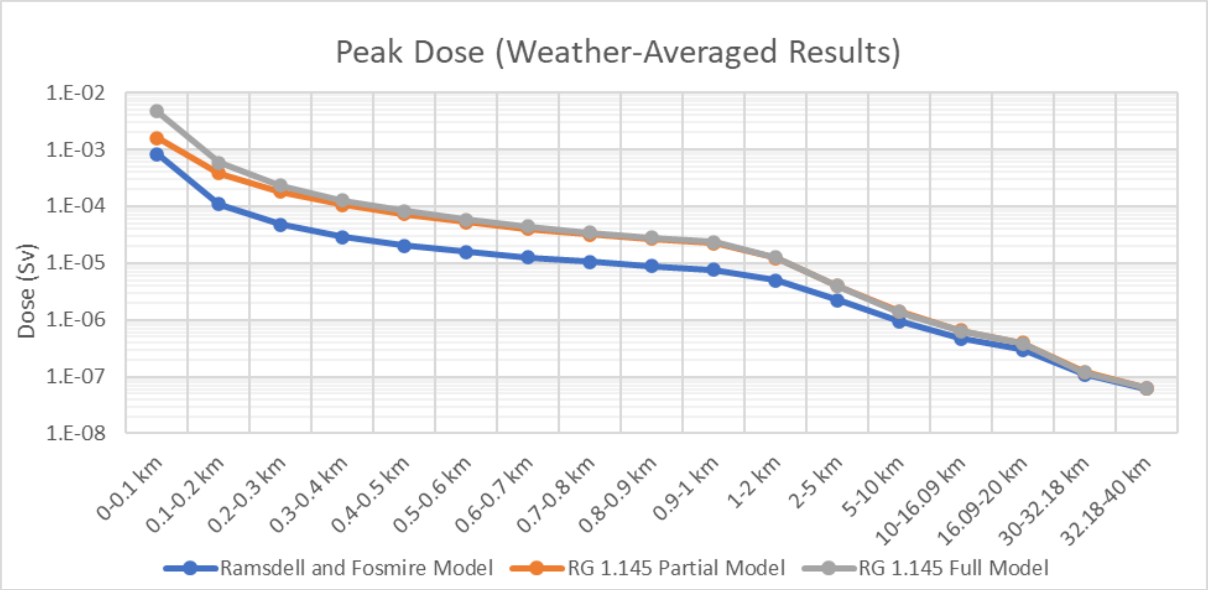
*Note: Consequence results are shown to illustrate code capabilities only. Actual consequences would be based on design, site, and scenario-specific factors.

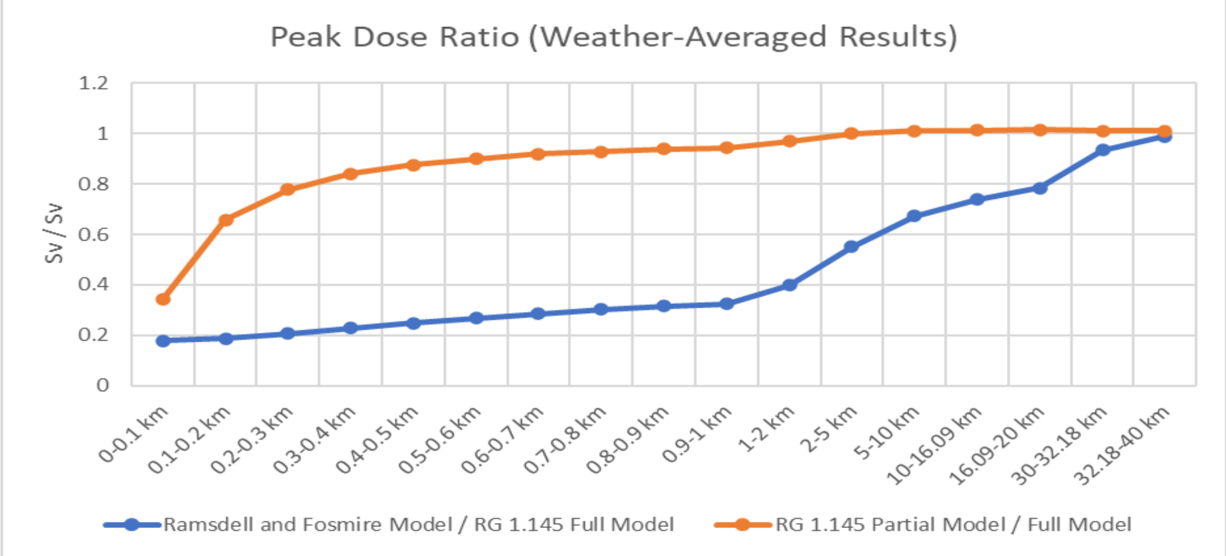
Sensitivity Analysis of Nearfield Modeling

- Using the weather conditions of the Sequoyah Nuclear Plant, the project compared the following three MACCS nearfield modeling approaches:
  (1) Regulatory Guide (RG) 1.145 Partial Model (with Area Source)
  (2) RG 1.145 Full Model (with Point Source)
  (3) Ramsdell and Fosmire Model (with Point Source)
- Both options 1 and 2 are based on the nearfield modeling approach described in RG 1.145.
  - Option 1 is a partial implementation of RG 1.145. This model does not directly account for building wake. Instead, the project uses an area source based on the building size to model the building wake zone.
  - Option 2 is a full implementation of RG 1.145. This modeling approach considers the effects of both building wake mixing and ambient plume meander.
- Option 3 is based on the Ramsdell and Fosmire nearfield modeling approach used by ARCON96. (SAND2021-6924)
- Options 2 and 3 are new models available in MACCS version 4.1

Sensitivity Analysis of Nearfield Modeling*

- The amount of plume spread in the different models notably impacts doses in the first 40 km (25 mi).
- Because the Ramsdell and Fosmire Model has the most spread, it has the lowest peak doses of the three nearfield models.
- The two RG 1.145 models have similar peak doses after roughly 1 km (0.62 mi), whereas the Ramsdell and Fosmire Model does not have the same peak doses until approximately 40 km (25 mi).

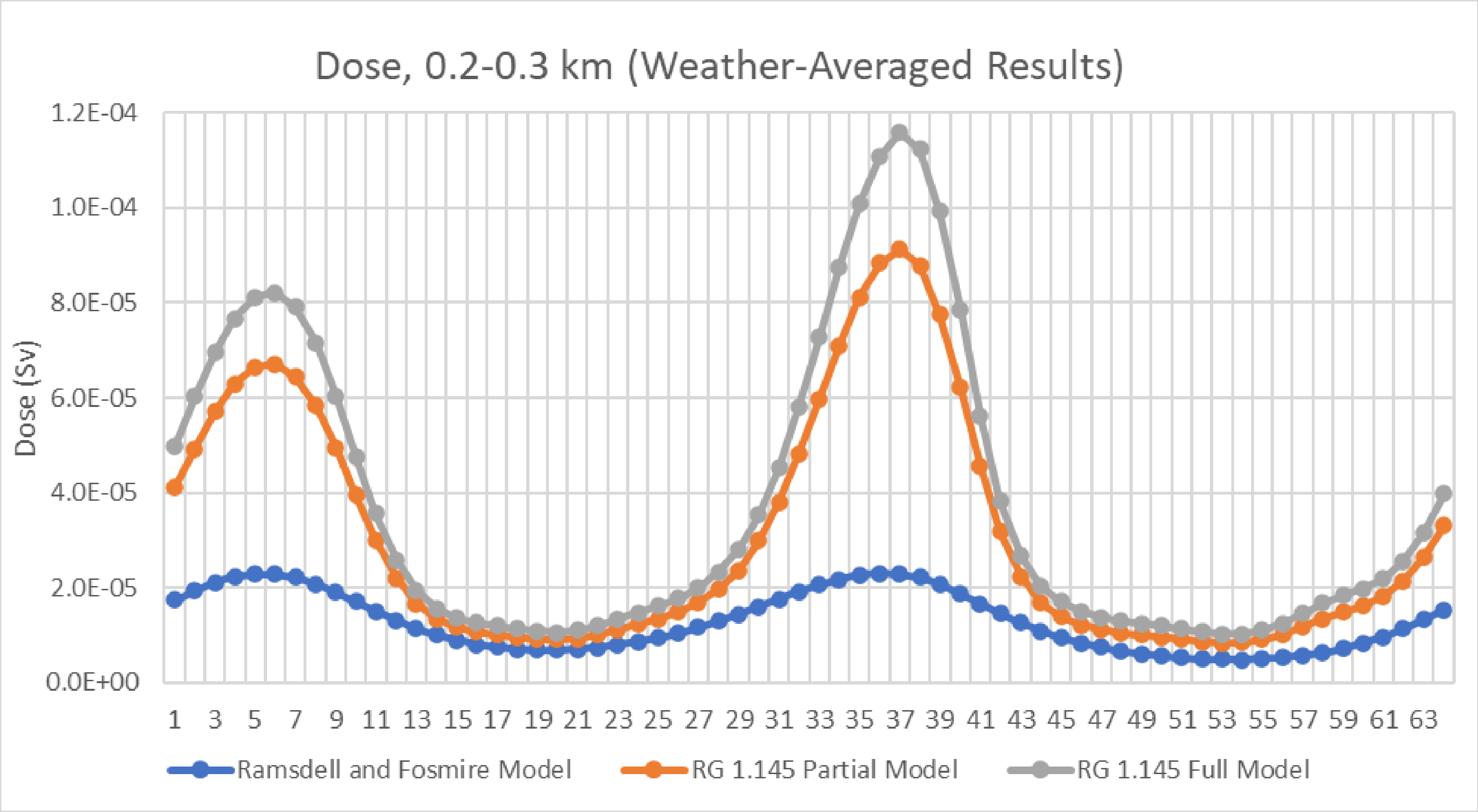




*Note: Consequence results are shown to illustrate code capabilities only. Actual consequences would be based on design and site-specific factors.

Sensitivity Analysis of Nearfield Modeling*

- There are 64 compass directions in the analysis. Direction 1 represents north, and ascending numbers represent a clockwise direction.
- The double peak results are due to the meteorological conditions at the Sequoyah Nuclear Plant site, which has two dominant wind directions.
- The RG 1.145 Full Model has the narrowest plume, and therefore it has the highest doses in the dominant wind directions.
- The Ramsdell and Fosmire Model has more horizontal plume spread, creating a more uniform dose.
- The Ramsdell and Fosmire Model shows a lower dose in all directions, likely because of vertical plume spread.



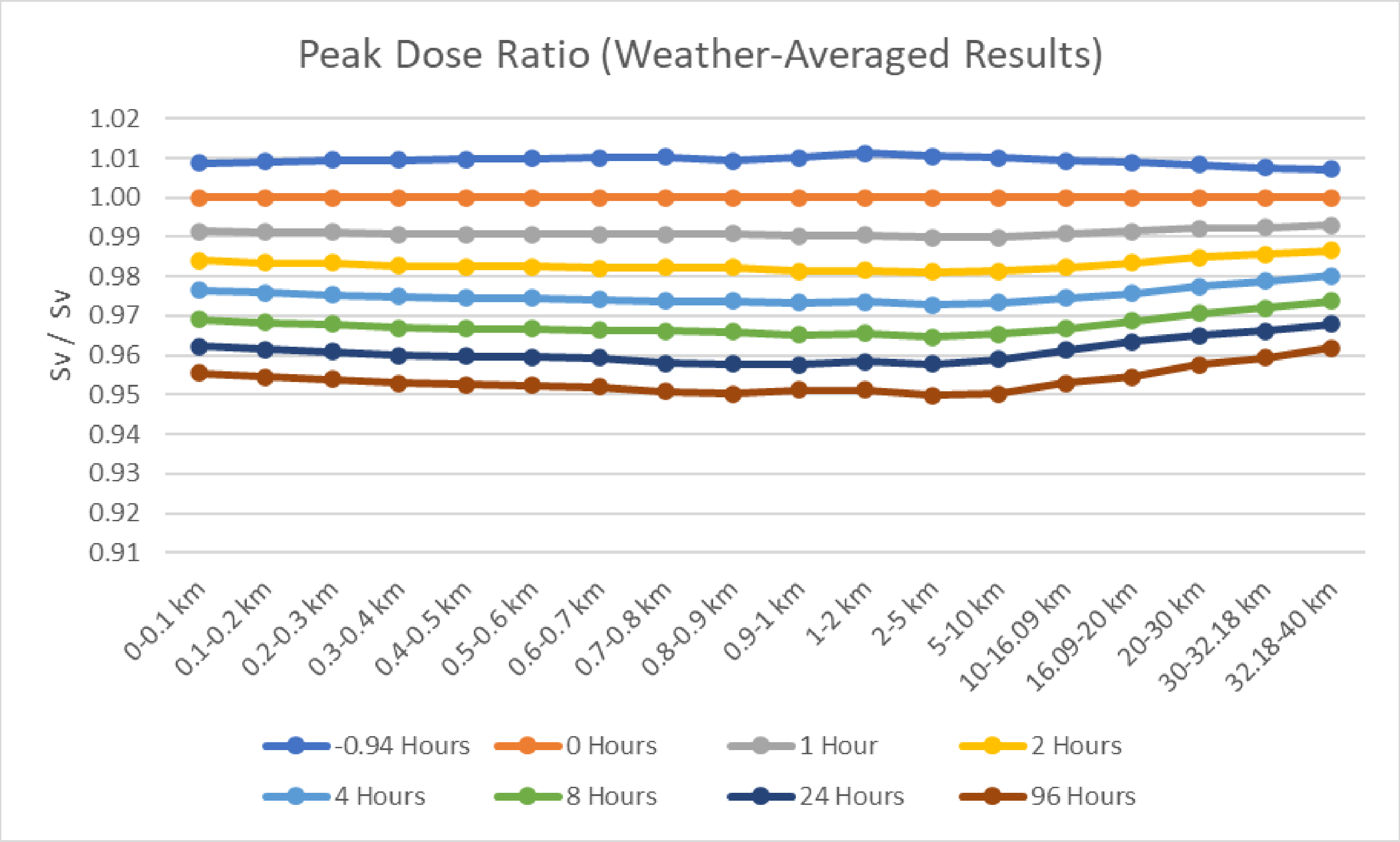
*Note: Consequence results are shown to illustrate code capabilities only. Actual consequences would be based on design, site, and scenario-specific factors.

Sensitivity Analysis of Release Timing

- MACCS assumes that reactor shutdown occurs at the beginning of an accident scenario. MACCS begins calculating decay and ingrowth at this time.
- The TOP scenario is different in that shutdown does not immediately occur.
- The reactor is postulated to operate for roughly an hour when the reactor power level steadily increases. This presents a few issues in computing offsite consequences:
  - Since reactor shutdown does not immediately occur, the holdup time between reactor shutdown and the start of release is shorter than MACCS anticipates.
  - The calculation of the core inventory assumes steady-state operation. If the reactor power level changes, it may not fully represent the new composition from the shift in fission rate before shutdown.
  - Release can begin before reactor shutdown. MACCS is designed only to calculate decay and ingrowth from a core inventory given at a fixed time. It does not model production of fission and activation products during release.

Sensitivity Analysis of Release Timing

- A shift in the release timing impacts the dose. This sensitivity evaluates the dose over a 7-day exposure period.
- The TOP release has a start time of 0.94 hours; a shift of -0.94 hours represents an immediate release.
- Despite a shift by up to 4 days, the projected dose remained within about 6 percent.
- Nearly half of this range (3 percent) can be attributed to a change in release timing of just 3 hours (i.e., from -0.94 to 2 hours).



Note: Consequence results are shown to illustrate code capabilities only. Actual consequences would be based on design and site-specific factors.

Summary Results

- The MELCOR analysis selected accident conditions during a TOP accident scenario that produced an example source term of the INL Design A conceptual design.
- The sensitivity analysis of the nearfield models using the weather conditions of the Sequoyah Nuclear Plant site shows the following:
  - The RG 1.145 Full Model has the highest peak dose.
  - The Ramsdell and Fosmire Model has the most plume spread.
  - The two RG 1.145 Models align quickly after 1 km.
  - The Ramsdell and Fosmire Model may not align the RG 1.145 Models until 40 km (25 mi).
- The other sensitivity analyses show the following:
  - Exposure to ground contamination over the long term is the dominant contributor to the overall lifetime dose projection in this example.
  - The use of an expanded subset of radionuclides in this example consequence analysis showed only a minimal increase in consequence.
  - A shift in the release timing for this example source term has a small impact on the projected peak dose.

Conclusions

- The results of our evaluation confirm that, despite some limitations, analysts can use the flexibility of the MACCS code to analyze the offsite consequences of an advanced reactor design under a postulated accident scenario.
- The evaluation exercise provided valuable practical experience in implementing new ORIGEN inventories and MELCOR source terms in MACCS.
- As new source terms of new and advanced reactor designs become available, RES staff may assess whether further enhancements to the MACCS code are needed.
- The project has identified several candidate future research activities:
  - continue to demonstrate MACCS capabilities using as input the core radionuclide inventory and atmospheric release from the example SCALE and MELCOR demonstration calculations,
  - continue the evaluation of radionuclides in non-LWR inventories important to dose and expanding these evaluations to include ingestion doses,
  - develop a method to analyze or conservatively bound accidents with simultaneous release and fission, and
  - develop methods to analyze or conservatively bound the impact of additional radionuclide chemical and physical forms and how they may transform in the environment.

Questions?

- A. Nosek, “MACCS Consequence Analysis Demonstration Calculations for an Example Heat Pipe Reactor Source Term,” U.S. Nuclear Regulatory Commission, March 2023, Agencywide Documents Access and Management System Accession No. ML23045A044
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- K. Wagner, C. Faucett, R. Schmidt, and D. Luxat, “MELCOR Accident Progression and Source Term Demonstration Calculations for a Heat Pipe Reactor,” SAND2022-2745, Sandia National Laboratories, Albuquerque, New Mexico, March 2022, ML22144A188.
- K.A. Clavier, D.J. Clayton, and C. Faucett, “Quantitative Assessment for Advanced Reactor Radioisotope Screening Utilizing a Heat Pipe Reactor Inventory,” SAND2022-12018, Sandia National Laboratories, Albuquerque, New Mexico, September 2022, ML22270A046.
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Licensing and Deployment Considerations for Factory-Fabricated Transportable Micro-Reactors

Advanced Reactor Stakeholders Meeting
July 20, 2023

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

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

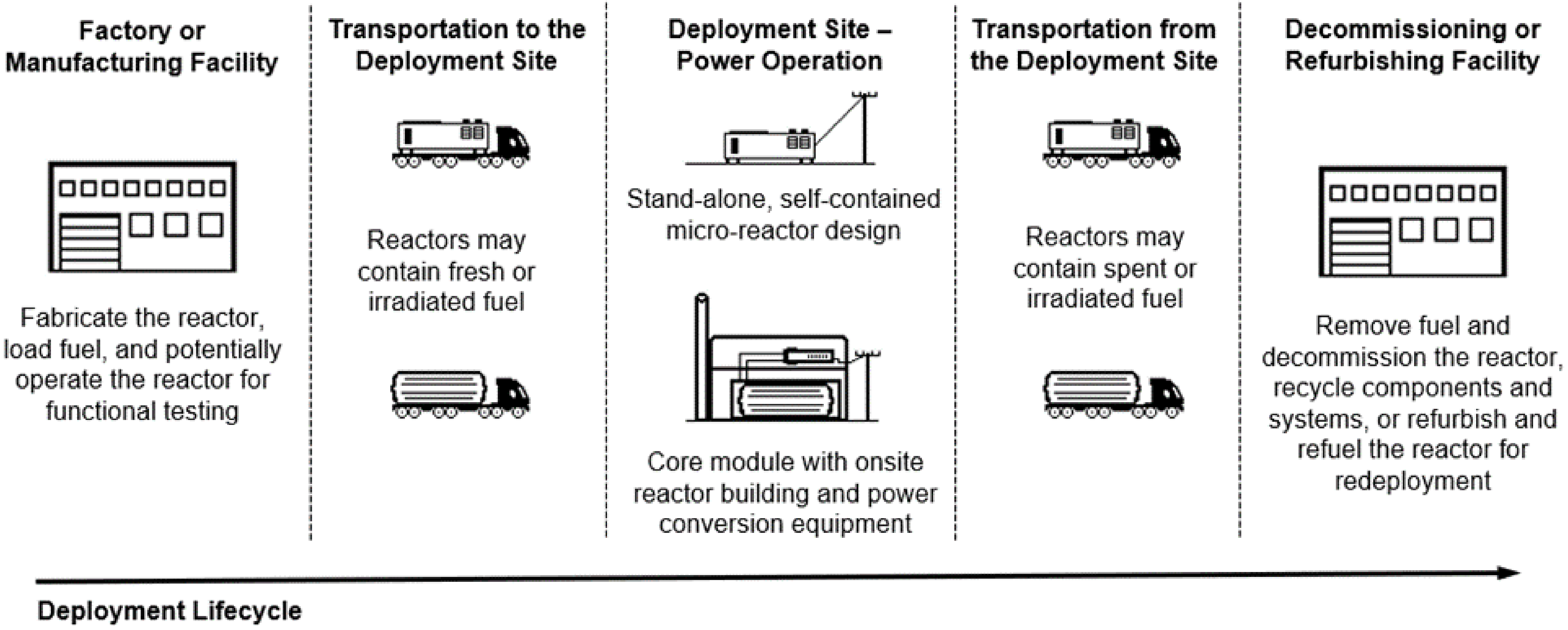
Contents

- Goals of this presentation
- Conceptual Deployment Model for Factory-Fabricated Transportable Micro-Reactors
- Regulatory Approaches for Fuel Loading at the Factory
- Regulatory Approaches for Operational Testing at the Factory
- Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps
- Next steps

Goals of this Presentation

- Inform stakeholders about regulatory approaches the NRC staff is developing for consideration by the Commission for fuel loading and operational testing at the factory
- Inform stakeholders about other licensing and deployment topics and potential near-term strategies and next steps the NRC staff is considering
- Receive feedback from stakeholders

Conceptual Deployment Model for Factory-Fabricated Transportable Micro-Reactors



Regulatory Approaches for Fuel Loading at the Factory

- The NRC staff is developing approaches for licensing fuel loading at the factory under the existing regulations for consideration by the Commission:
  – Facility operating license issued pursuant to 10 CFR Part 50 that limits operation to fuel loading
  – Combined license issued pursuant to 10 CFR Part 52 that limits operation to fuel loading
  – Manufacturing license for manufacture and possession of the utilization facilities and a license to possess special nuclear material issued pursuant to 10 CFR Part 70 with provisions for the utilization facilities to include features to preclude criticality

Regulatory Approaches for Operational Testing at the Factory

- The NRC staff is developing approaches for licensing operational testing at the factory under the existing regulations for consideration by the Commission:
  - Construction permit issued pursuant to 10 CFR Part 50, potentially covering many reactors, that would be converted to 10 CFR Part 50 facility operating licenses that limit operation to that needed for operational testing
  - Combined licenses issued pursuant to 10 CFR Part 52, potentially issued at the same time based on one application, that limit operation to that needed for operational testing
  - Construction permit issued pursuant to 10 CFR Part 50, potentially for many *commercial non-power reactors,* that would be converted to facility operating licenses that limit operation to that needed for operational testing

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Considerations related to initial fuel load and authorization to operate at the deployment site for reactors that arrive pre-loaded with fuel

- Deployment strategies that include loading fuel or operational testing at a manufacturing facility would result in fueled reactors arriving at the deployment site
- Several requirements in the Atomic Energy Act of 1954, as amended (AEA), and 10 CFR Parts 50 and 52 that are related to public notifications, the opportunity for hearing, authorization to operate the facility, and others are premised on fuel being initially loaded at the deployment site
- The NRC staff is considering whether there is a suitable alternative to “initial loading of fuel” at the deployment site that could be used as an alternate milestone and would accomplish the underlying purpose of the AEA and regulations

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Timeframe for authorization to operate at the deployment site

- Factory-fabricated transportable micro-reactors may have significantly simpler and shorter construction activities at the deployment site compared to large light water reactors and could be ready to begin operation in days to weeks to a few months after obtaining a construction permit or combined license
- Several requirements in the AEA and 10 CFR Part 50 and Part 52 that are related to the environmental review, the schedule for intended operation, public notifications, the opportunities for hearing, authorization to operate the facility, and others include timeframes that could add up to many months in total
- For licensing under 10 CFR Part 52, the NRC staff plans to clarify the circumstances under which the schedule for intended operation and initial fuel load can be accelerated and is considering ways to streamline public notifications, hearings, and the authorization to operate, as appropriate
- For licensing under 10 CFR Part 50, the NRC staff is considering opportunities to expedite steps in the processing and review of applications for facility operating licenses, such as acceptance review and docketing, milestones for hearings, and the supplement to the environmental impact statement

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Licensing replacement reactors

- Factory-fabricated transportable micro-reactors might be periodically replaced with reactors of the same design at the end of their lives or fuel cycles, and each reactor would be required to have its own combined license or facility operating license
- A licensee might have multiple fueled reactors on site in various states of operation and shutdown to allow for transition from the operating reactor to the replacement reactor with minimal downtime. This would need to be considered in the safety and environmental reviews
- The NRC staff previously addressed similar concepts and considered licensing options for multi-module facilities in SECY-11-0079, "License Structure for Multi-module Facilities Related to Small Modular Nuclear Power Reactors," dated June 12, 2011 (ADAMS Accession No. ML110620459)
- The NRC staff is considering approaches under 10 CFR Part 50 and Part 52 where the construction permit application or combined license application would cover all reactors envisioned to be operated at the deployment site and each reactor would be authorized to begin operation under its own facility operating license or combined license once the Commission had made the required findings

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Autonomous and remote operations

- Proposed designs for factory-fabricated transportable micro-reactors (and potential designs for other types of reactors) might include autonomous and remote operational characteristics to reduce the number of operators and other categories of personnel at the facility site
- As previously noted in SECY-20-0093, “Policy and Licensing Considerations Related to Micro-Reactors,” dated October 6, 2020 (ADAMS Accession No. ML20129J985), both autonomous and remote operations raise potential policy-related matters
- The NRC staff plans to further develop its understanding of the industry deployment models for factory-fabricated transportable micro-reactors with respect to industry plans for remote and autonomous operations, identify any gaps in the existing human factors engineering review needed to address the deployment models, and develop the technical bases for any new guidance that may be needed
- As part of the proposed Part 53 rulemaking provided to the Commission, the NRC staff has proposed a new risk-informed, performance-based, technology-inclusive cybersecurity framework that would require licensees to demonstrate protection against cyberattacks in a manner that is commensurate with the potential consequences from those attacks

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Transportation of fueled reactors

- Factory-fabricated transportable micro-reactor developers (and potentially developers of floating nuclear power plants that use reactors with higher power levels) envision transporting fueled reactors from a fabrication site or a refurbishment and refueling facility to the deployment site for operation and later removing fueled reactors from the deployment site at the end of their useful lives or fuel cycles
- Transportation packages for factory-fabricated transportable micro-reactors may consist of the reactor itself or the reactor plus additional overpack, as needed. Packages for transporting a micro-reactor from the factory to the deployment site could be either a Type A fissile (Type AF) or Type B fissile (Type BF) package, as defined in 10 CFR Part 71
- The NRC staff intends to use the existing regulatory framework (primarily 10 CFR Part 71) to review transportation of fueled commercial micro-reactors in the near term, which may include the use of the alternate test criteria in 10 CFR 71.41(c), the special package authorization option in 10 CFR 71.41(d), or exemptions, as appropriate

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Storage of fuel after irradiation in a power reactor

- Depending on the duration between withdrawal of the fuel from the reactor (or the final reactor shutdown) and placement into a dry storage facility, different regulations may apply to the storage of the reactor fuel
- The definition of spent fuel in 10 CFR 72.3 includes criteria that the fuel has been withdrawn from a nuclear reactor following irradiation and has undergone at least one year's decay since being used as a source of energy in a power reactor
- In order to store irradiated power reactor fuel that had been withdrawn from a reactor for less than a year in an independent spent fuel storage installation, the licensee would be required to apply for a specific license under 10 CFR Part 72 and request and justify exemptions addressing the one-year decay time requirement in the regulations
- The NRC staff intends to engage with stakeholders as they further develop their strategies for handling and storage of irradiated and spent fuel generated in factory-fabricated transportable micro-reactors

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Decommissioning process and decommissioning funding assurance

- Factory-fabricated transportable micro-reactor deployment models might involve transporting a reactor away from the deployment site to a facility at a different location for decommissioning at the end of its life or for refurbishment and refueling before re-deployment
- Depending on the activities to be conducted at a decommissioning facility or a refurbishment and refueling facility, the facility may need to be licensed under a combination of the regulations in 10 CFR Part 30 for byproduct material, Part 50 or 52 for a facility operating license or combined license, Part 70 for special nuclear material, and Part 72 for spent fuel storage
- The deployment site licensee would need to establish decommissioning funding assurance that considers the cost of removing the reactor from the site and decommissioning it elsewhere in addition to the cost of decommissioning activities at the deployment site. The NRC staff may consider site-specific decommissioning cost estimates that appropriately account for all activities at both locations and all waste disposal costs

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Siting in densely populated areas

- Some micro-reactor license applicants might seek to site reactors at locations that are inconsistent with the current Commission policy and the regulations in 10 CFR 100.21(b), i.e., a location within a population center of 25,000 residents or more
- The NRC staff is currently revising the population-related siting guidance in Regulatory Guide (RG) 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Revision 3, issued March 2014 (ADAMS Accession No. ML12188A053) to provide technology-inclusive, risk-informed, and performance-based criteria to assess certain population-related issues in siting advanced reactors
- In the near term, the staff will continue its effort to revise RG 4.7 and will review license applications in accordance with current Commission policy that allows alternative population-related criteria but precludes siting a commercial power reactor, no matter the size or type of reactor, within a population center of 25,000 residents or more

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Commercial maritime applications

- The NRC staff is aware of growing interest in commercial maritime applications of factory-fabricated transportable micro-reactors and other reactor technologies for stationary power production, marine vessel propulsion, production of decarbonized fuels, and other uses
- Depending on the particular application, deployment of commercial maritime reactors could introduce a host of policy issues and legal matters, especially for nuclear propulsion in the international shipping industry
- The NRC staff will continue to engage with stakeholders and monitor developments related to commercial maritime applications and assess the need for future Commission direction

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Commercial space applications

- The NRC staff is aware that developers are considering space applications of factory-fabricated transportable micro-reactors. However, the NRC staff is not aware of any plans for fully commercial space applications
- In the case of a fully commercial space application of a factory-fabricated transportable micro-reactor, the NRC's established regulatory jurisdiction and licensing authority would cover the related terrestrial activities prior to launch activities, which would be under the authority of the Federal Aviation Administration's Office of Commercial Space Transportation (a part of the Department of Transportation)
- If developers engage the NRC staff on terrestrial activities related to commercial space applications of factory-fabricated transportable micro-reactors, the NRC staff intends to apply the established regulatory framework, as informed by the potential licensing approaches and strategies outlined in this presentation

Other Licensing and Deployment Topics and Potential Near-Term Strategies and Next Steps

Commercial mobile applications

- Factory-fabricated transportable micro-reactor deployment models might include scenarios where the reactor would be operated on an as-needed, where-needed basis, such as for disaster relief or to meet temporary increases in demand
- The current regulatory framework for reactor licensing is not conducive to this deployment strategy because the regulations in 10 CFR Part 100 apply to every site at which a reactor may be operated, and NRC's implementation of the National Environmental Policy Act relies on performing an environmental review that contemplates a particular site
- The AEA and regulations in 10 CFR Parts 50 and 52 for licensing utilization facilities also require opportunities for public hearings before the Commission can issue a facility operating license or authorize operation under a combined license. These may take a minimum of several months to complete, limiting the ability to rapidly deploy a reactor to meet immediate, short-term needs
- The NRC staff will monitor developments in the commercial sector related to deployment models and the demand for commercial mobile micro-reactor licensing. The staff will assess the need for future Commission direction, rulemaking, and coordination with other Federal agencies in this area

Next Steps

- Publish a draft white paper to further stakeholder engagement in August 2023
- Develop a Commission paper on licensing and deployment considerations factory-fabricated transportable micro-reactors:
  - Request Commission direction on regulatory approaches for loading fuel and operational testing at the factory
  - Provide information on other topics, including the NRC staff's related near-term strategies and next steps

Discussion Items

- Are there other approaches that the NRC staff should consider for loading fuel and operational testing at the factory that would not involve rulemaking?
- Are there other near-term strategies the NRC staff should consider for the other identified topics?
- Other feedback or questions for the NRC staff.

Break

Meeting will resume at 4:30 pm EST

Microsoft Teams Meeting

Bridgeline: 301-576-2978

Conference ID: 501 432 683#

Transportable Microreactor Package Approval

Advanced Reactor Stakeholder Meeting
July 20, 2023

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

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

Purpose

- Inform external stakeholders of the NRC activities related to transportable microreactors.
- Importance and benefit of timely pre-application engagements in the regulatory process.

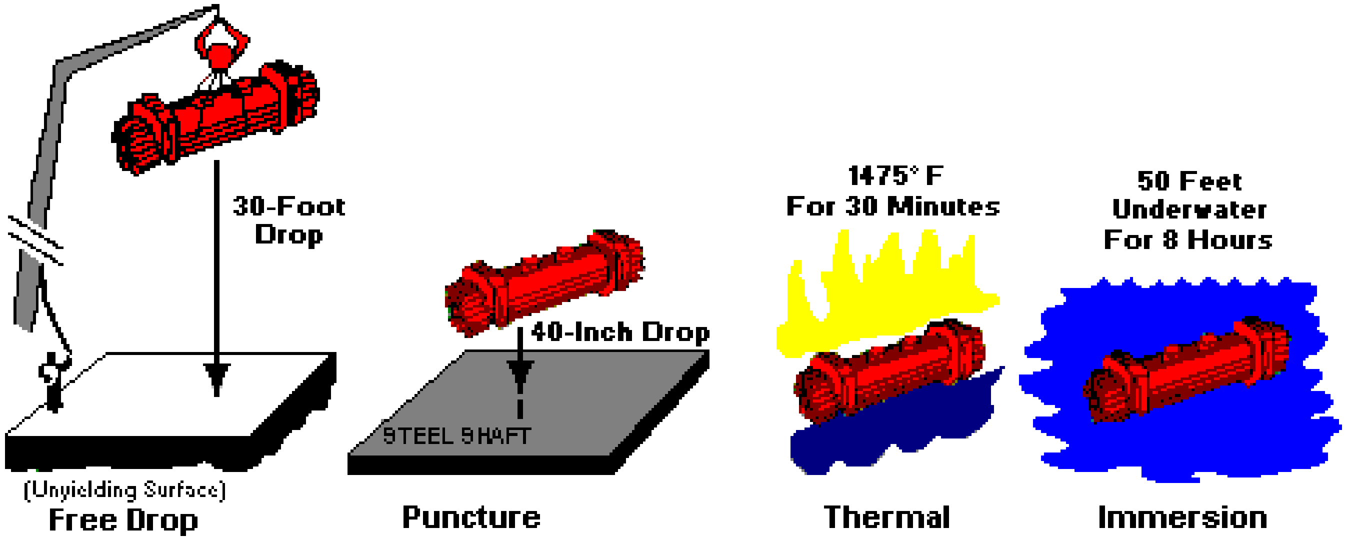
Overview

- Package approval standards
- Package approval regulatory approaches
- Microreactor Package Approval
- Risk-Informed Methodology

Package Approval Standards

- Prescriptive performance-based regulations
- Tests and conditions
  - Normal conditions of transport
  - Hypothetical accident conditions
- Post-test criteria
  - Criticality safety
  - Maximum dose rates
  - Containment criteria

Package Tests



- Normal conditions of transport (10 CFR 71.71 & 49 CFR 173.465)
  - Hot and cold temperatures
  - Reduced and increased external pressure
  - Vibration
  - Water spray
  - Free drop (1 foot)
  - Penetration test
- Hypothetical accident conditions (10 CFR 71.73)
  - 30-foot drop test in most damaging orientation
  - 40-inch puncture test
  - 30-minute fire test
  - Water immersion test

Post-Test Performance Criteria

- Criticality safety
  - Single package (10 CFR 71.55)
  - Array of packages (10 CFR 71.59)
- Maximum dose rates for normal transport

  (10 CFR 71.47 & 49 CFR 173.441)
- Additional criteria for Type B packages (10 CFR 71.51)
  - Containment for normal conditions of transport and hypothetical accident conditions
  - Maximum dose rates after hypothetical accident conditions

Package Approval Regulatory Approaches

- Final Draft: Micro-reactors Licensing Strategies (ML21328A819)
- 10 CFR 71.41(c)
  - Limited to changes to tests for normal conditions of transport and hypothetical accident conditions
  - No changes to acceptance criteria
  - Shipper controls for equivalent level of safety
- 10 CFR 71.41(d)
  - One-time shipment of large packages
  - Special package authorization
  - Equivalent level of safety
- Exemptions via 10 CFR 71.12

Microreactor Package Pre-Application Engagements

- Provide staff with knowledge on specific designs and technologies
- Enhances quality of applications
- Helps NRC to understand future needs and inform its budget
- Ensures applicants and regulator have shared understanding of
  - the applicable requirements
  - review approach and
  - whether data gaps exist (e.g., testing) that need to be addressed, as these may be the critical path, impacting the overall schedule
- Allows NRC to plan for package reviews

Risk-Informed Methodology

- NRC is reviewing a risk-informed methodology (ML23066A201) for limited number of shipments for a single transportable microreactor.
- Staff can approve exemptions that meet criteria in §71.12, unless directed to send to Commission.
- Uncertainties on the use of risk-informed methodology:
  - May only be used by one reactor vendor
  - Number and type of exemptions requested for a transportable microreactor package approval
- Planning a Commission paper on risk-informed methodology.
- Significant number of transportable microreactor package approvals needing exemptions would likely warrant Commission direction.

Closing Remarks

- NRC is ready to review packages for transportable microreactors.
- NRC regulatory framework in 10 CFR Part 71 is adequate for approving transportable microreactors.
- Package approval method could be package/reactor dependent.
- NRC is aware of numerous transportable microreactor designs but has not had pre-application engagement on most of them.

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Future Meeting Planning

- The next periodic stakeholder meetings in 2023 are scheduled for September 14, October 25, and December 7.

- 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=20230270

- Then, click link to NRC public feedback form:

Meeting Feedback

Meeting Feedback Form EXIT

Meeting Dates and Times