

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	S NRC						
10:45 am – 11:00 am	00 am Introduction to the New Fuels Atlas							
11:00 pm – 11:30 am	Update on SCALE/MELCOR Non-LWR Source Term and Fuel Cycle Demonstration Project	n 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



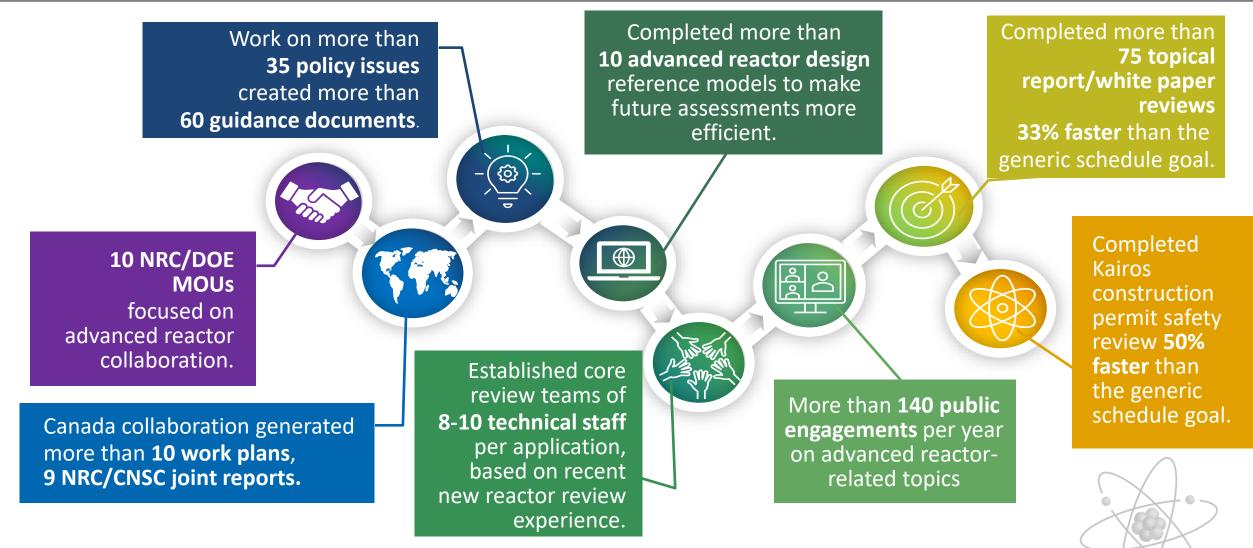


- Interim report available through link on: <u>https://www.nrc.gov/reactors/new-</u> <u>reactors/advanced/international-</u> <u>cooperation/collaboration-with-canada.html</u>
- Also available in NRC ADAMS at ML23172A201
- Feedback is welcome, especially related to enhancing usefulness for potential license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

Contact: <u>Steve.Jones@nrc.gov</u> or <u>Jorge.Hernandez@nrc.gov</u>

NRC's Advanced Reactor Readiness By the Numbers

Statistics since 2018



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

5	trategy 1 Knowledge, Skills, and Capability								l	Leg	end																	
5	trategy 2 Computer Codes and Review Tools		Concurr	ence (Div	rision	/Inte	eroffi	ice)				•	ED	00	oncu	rren	ce P	erio	bd									
\$	trategy 3 Guidance		Federal	Register	Publi	icati	on						Co	mmi	ssion	n Re	view	Per	riod	**								
	trategy 4 Consensus Codes and Standards		Public C	Comment F	Perio	d							AC	RS :	SC/F	C (S	Sche	dule	ed or	r Plai	nned)						
\$	trategy 5 Policy and Key Technical Issues		Draft Iss	suance of	Deliv	vera	ble						Ext	terna	al Sta	ikeh	olde	r Inte	erac	tions	3							
\$	trategy 6 Communication		Final Iss	suance of	Deliv	vera	ble					1	Pu	blic I	Meeti	ing (Sch	edul	led o	or Pla	anneo	d)					Versio	on
														Pre	esen	t Da	ay										7/11/2	23
		0										2022)				Í						202	8				
Strategy	Regulatory Activity	Commission Papers	Guidance	Rulemaking	NEIMA	Complete	Jan	Feb	Mar	Apr			Aug	Sep	Oct	Nov	Dec	Jan	Feb	Mar	Apr	Τ		Π	Aug	Sen	Nov	Dec
	Development of non-Light Water Reactor (LWR) Training for Advanced Reactors (Adv. Rxs) (NEIMA Section 103(a)(5))					x																						
1	FAST Reactor Technology				Х	Х																						
1	High Temperature Gas-cooled Reactor (HTGR) Technology				Х	Х																						
	Molten Salt Reactor (MSR) Technology				Х	Х																						
	Competency Modeling to ensure adequate workforce skillset					х																						l i
	Identification and Assessment of Available Codes					Х																	\top	Π				\Box
	Development of Non-LWR Computer Models and Analytical Tools																											
	Code Assessment Reports Volume 1 (Systems Analysis)					Х																					\square	
	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																		Π				\Box
	Reference plant model for Monolith-type Micro- Reactor																											\square
	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																\square		П				
	FAST code assessment for TRISO fuel					х																						Li
	Code Assessment Reports Volume 3 (Source Term					Х																						
	Non-LWR MELCOR (Source Term) Demonstration Project					x								Ļ								T						
	Reference SCALE/MELCOR plant model for					~																\top	\top	H			1	\square



https://www.nrc.gov/reactors/new-reactors/advanced/integrated-review-schedule.html 7



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



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)

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)

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)

CatEx – Category Exclusion

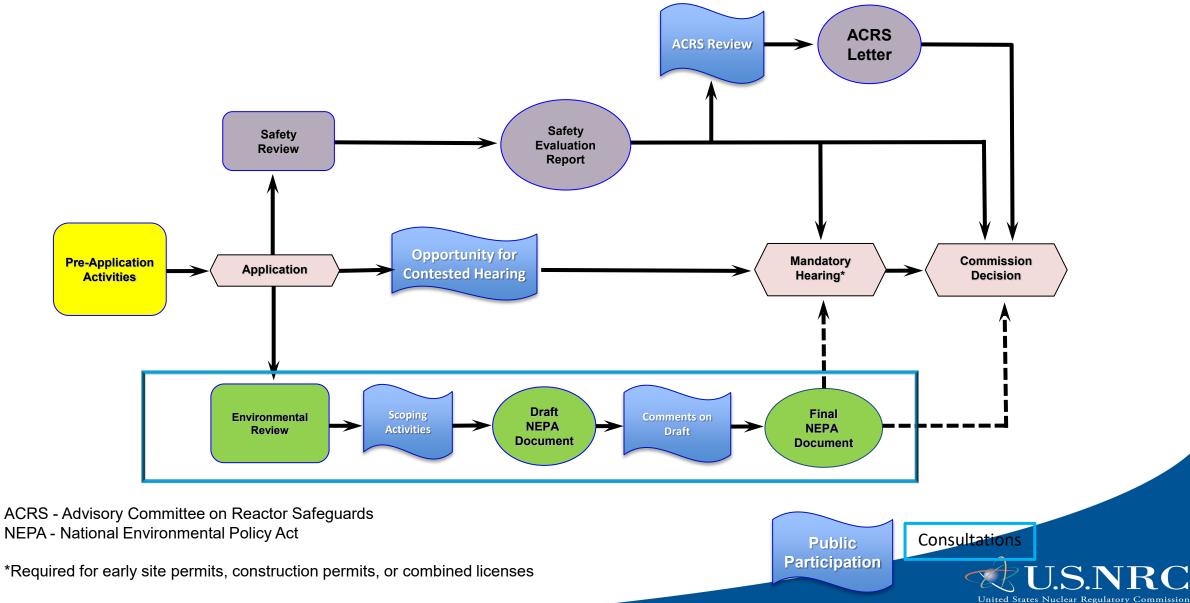
- EA Environmental Assessment
- FONSI Finding of No Significant Impact
- EIS Environmental Impact Statement
- ROD Record of Decision





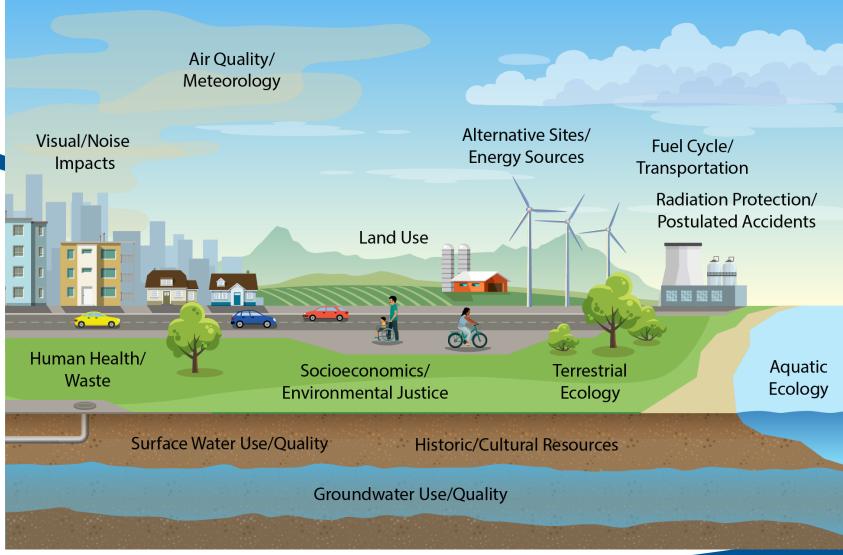
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Licensing Process and the Environmental Review



Protecting People and the Environment

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

Coordination with Federal, State, local, and Tribal **Government Agencies** At. NOISS GHIND



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 (<u>ML19190A078</u>)
 - Feedback from Public Meetings (ANR Stakeholder, Scoping, Draft EIS meetings)
 - Regulatory Information Conference (RIC) Sessions in 2020 (<u>RIC 2020</u>), 2021 (<u>RIC 2021</u>), and 2023 (<u>RIC 2023 Environmental</u> and <u>RIC 2023 Siting</u>)
 - 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) (<u>ML21222A044</u>)
 - 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 (<u>Law</u>)
 - Advance Act (<u>Proposal</u>)
 - FRA (<u>Law</u>)
 - FAST-41 Implementation through Federal Permitting Improvement Steering Council (FPISC) (<u>Administration Website</u>)
 - Council on Environmental Quality (CEQ) Guidance Updates (FRN)
 - CEQ Regulatory Changes (<u>CEQ Website</u>)
 - 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 (<u>ML17312B567</u>) 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 <u>LIC 116</u>
- Expanded Use of Public Meetings and Regulatory Audits
 - NMSS follows NRR Office Instruction <u>LIC-111</u>
 - Optimization based on lessons learned
- Optimized use of Requests for Additional Information (RAIs)
 - NMSS follows NRR Office Instruction <u>LIC-115</u>
 - 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 (<u>Link</u>) 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 (<u>FRN</u>)
- 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 (<u>FRN</u>)
 - Fusion (<u>Website</u>)
 - Accident Tolerant Fuel (<u>Website</u>)



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

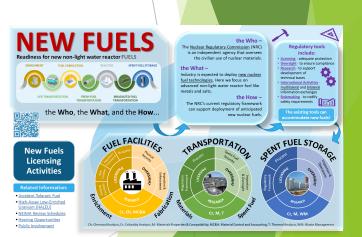
New Fuels Environment

- 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



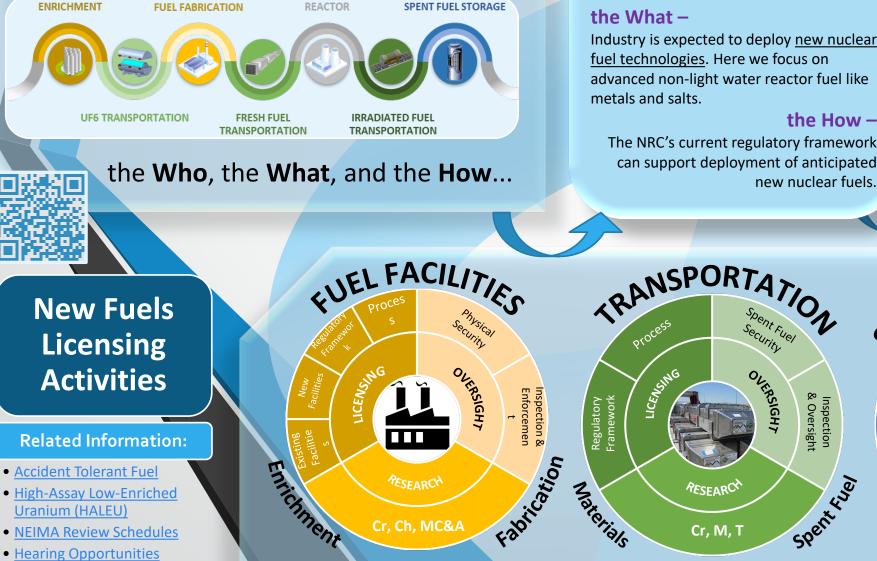
New Fuels Infographic

- 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

Public Involvement



the Who –

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

Industry is expected to deploy new nuclear advanced non-light water reactor fuel like

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

Inspection & Bations

OVERSIGHT

The existing tools can accommodate new fuels!

SPENT FUEL STOR

RESEARCH

Cr, M,

WM

Process

Regulatory Framework

LICENSI

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

New Fuels Website

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

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C 🗄 https://www.nrc.gov/ma	aterials/new-fuels.html	🗅 A" 🏠 🗲 😪
IUCLEAR REACTORS NUCLEAR MAT	TERIALS RADIOACTIVE WASTE NUCLEAR SECURITY PUBLIC MEETINGS & INVOLVE	MENT NRC LIBRARY ABOUT NRC
me Nuclear Materials		
≹ Navigation	New Fuels	Key Topics
ypes of Nuclear Materials	Traditional light-water reactor (LWR) fuel consists of uranium oxide fuel pellets enriched up	New Fuels Infographic
uel Cycle Facilities	to 5 weight percent uranium-235 (U-235), within an array of zirconium-based cladding. The nuclear industry is designing reactor fuels with operational conditions that differ significantly	New Fuels Licensing Activities
ledical, Industrial, & Academic Uses of uclear Materials	from traditional LWR fuels in that they may take different forms and include new materials. We recognize that there are considerations for these new fuel technologies across the entire nuclear fuel cycle, including enrichment, fuel fabrication, transportation, in-reactor	High-Assay Low-Enriched Uranium (HALEU)
laterials Transportation	requirements, reprocessing and spent fuel storage and disposal.	
ational Materials Program	The NRC licenses and regulates the Nation's civilian use of radioactive materials to provide	Related Information
polboxes	reasonable assurance of adequate protection of public health and <u>safety</u> and to promote the common defense and <u>security</u> and to protect the <u>environment</u> . The NRC considers <u>public</u>	Advanced Reactor Fuel
/orking Groups	<u>involvement</u> in, and information about, our activities to be a cornerstone of strong, fair regulation of the nuclear industry.	Accident Tolerant Fuel
Iranium Recovery	The NRC can review and regulate near-term uses of these fuel designs, including review of	ATF-Related Licensing Actions
lew Fuels	applications with the current regulatory framework. The Infographic below shows the regulatory tools we have in place to oversee an advanced fuel cycle. These tools include licensing, oversight, rulemaking, and research. The infographic is interactive with links to	
Fusion Energy Systems	the relevant programmatic areas.	

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

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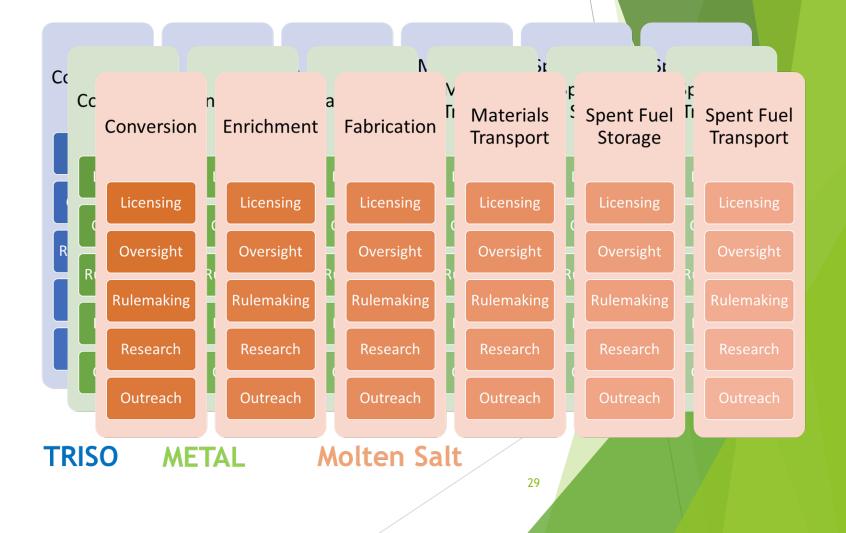
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The Regulatory Planner

- 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

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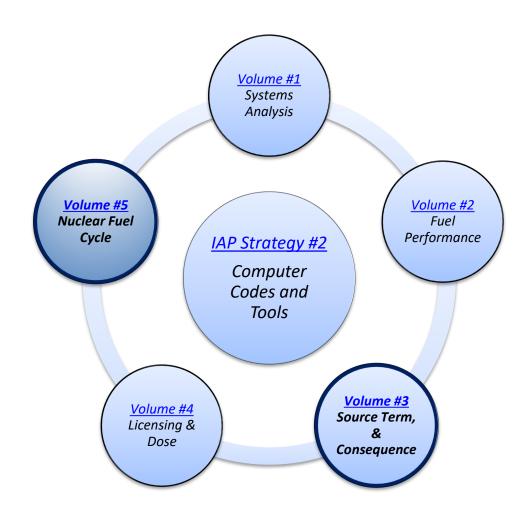
NRC's Strategy for Preparing for non-LWR Applications

• NRC's Readiness Strategy for Non-LWRs



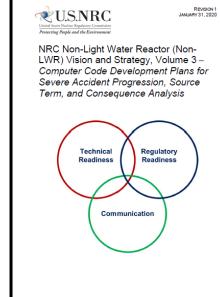
- IAPs are planning tools that describe:
 - Work, resources, and sequencing of work to achieve readiness

- 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



Volume #3



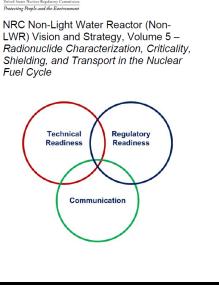
NRC's comprehensive neutronics package

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



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



WUSNRC

REVISION 1 MARCH 31, 2021

Volume #5

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



General Approach

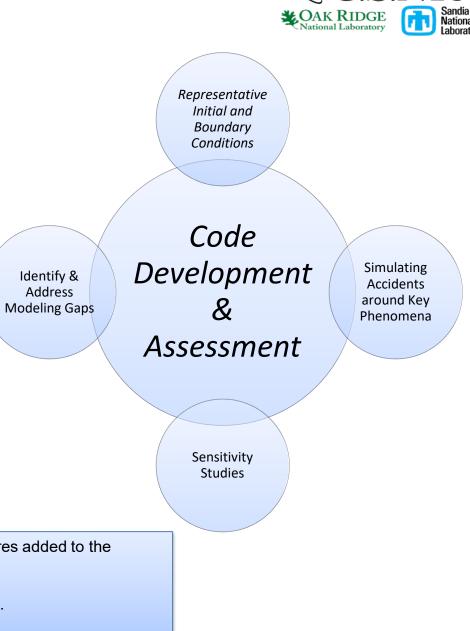


1. Build SCALE core models and MELCOR fullplant models

- 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

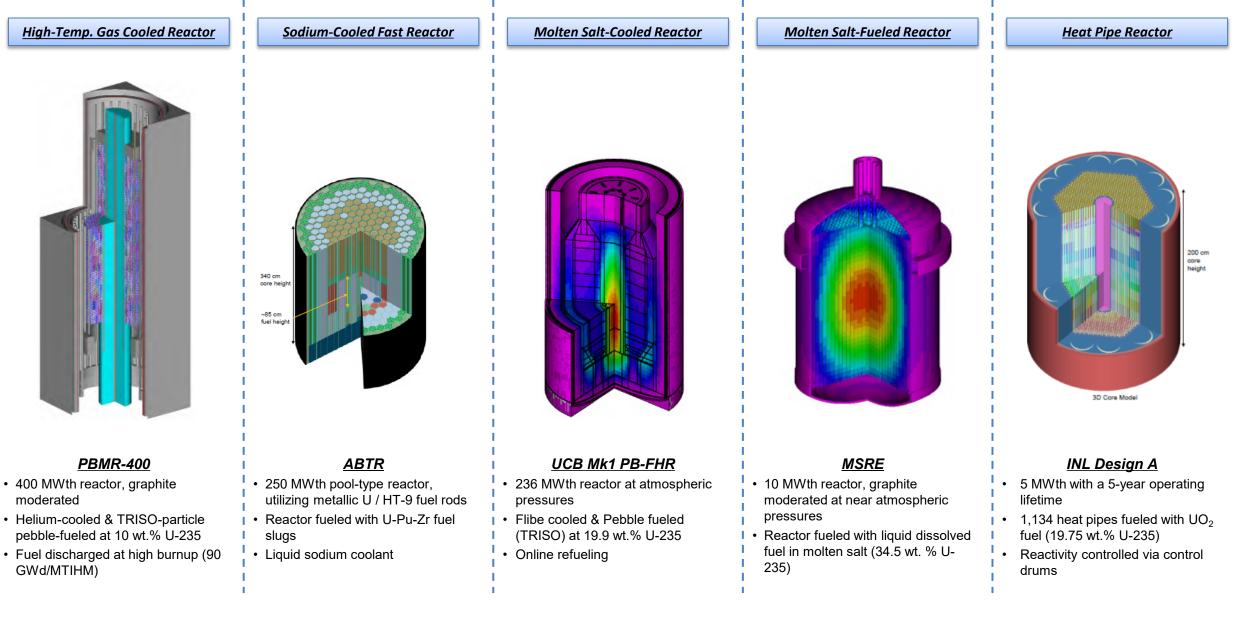
Source ren	n Demonstrati	onProject
	Molten Salt Fueled Reactor September 13, 2022	scale
	LCOR non-LWR source term demonstra	ation project
Heat-pipe reactor workshop on June 2: Slides @ Video Recording ram SCALE report @ MELCOR report @ MELCOR report @	9, 2021	June 29, 2021
High-temperature gas-cooled reactor w Sides g Video Recording III SCALE report g MELCOR report g	vorkshop on July 20, 2021	July 20, 2021
Fluoride-salt-cooled high-temperature i Slides Video Recording SCALE report MELCOR report	reactor workshop on September 14, 2021	September 14, 2021
Molten-salt-fueled reactor workshop or Slides g Video Recording (SCALE report g	n September 13, 2022	September 13, 2022
	on September 20, 2022	

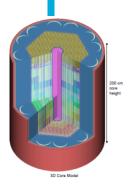
Public workshop videos, slides, reports at <u>advanced reactor source term webpage</u> SCALE input models available <u>here</u>. MELCOR input models available upon request.

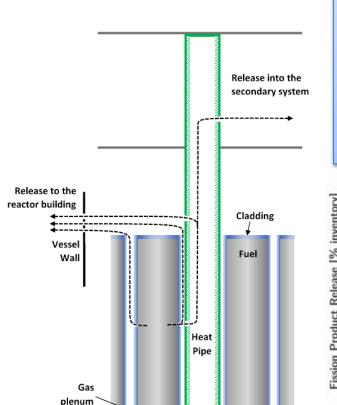


Non-LWR Designs Considered & Project Scope









Heat Pipe Reactor – INL Design A

Accidents Modeled

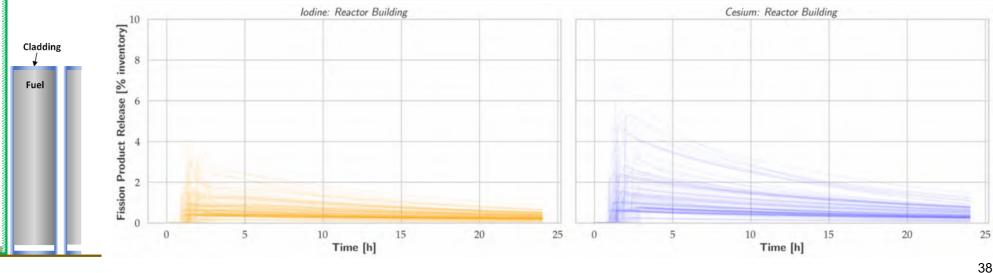
- Transient overpower
- Loss-of-heat sink
- Anticipated transient w/o SCRAM

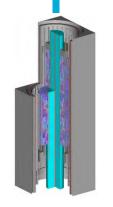
Insights Gained on BDBA Behavior

• Following scram, passive heat dissipation into reactor cavity ends the release from fuel

National

- 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.





Reactor equipment

maintenance and

repair building

Crane central room

Electrical-technical

building

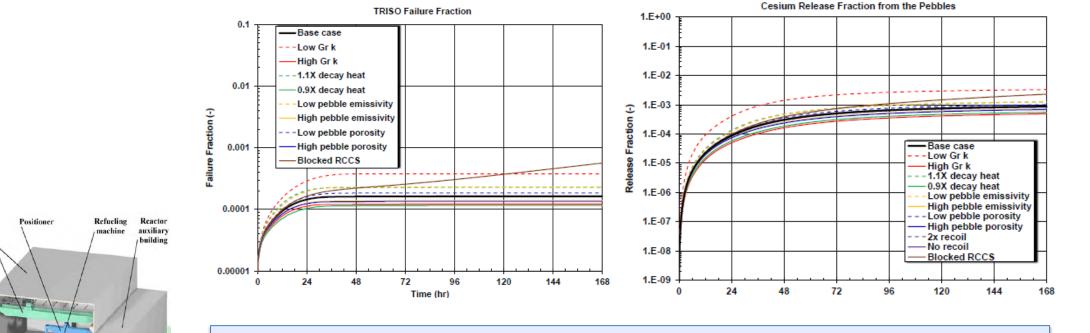
Power

conversion

system







Accidents Modeled

Depressurized loss-of-forced circulation

Insights Gained on BDBA Behavior

- 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

cavity

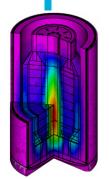
cooling

system

Reactor

Reactor Building

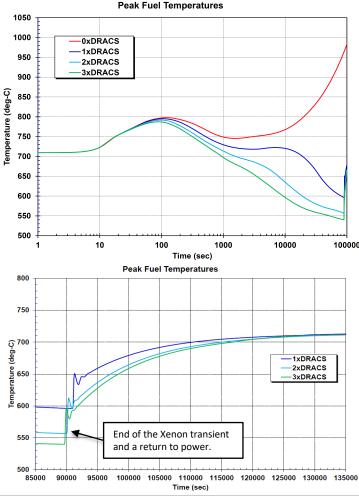
"HTGR Mechanistic Source Terms White Paper,' July 2010, [INL-EXT-10-17999]



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



ATWS with variable DRACS

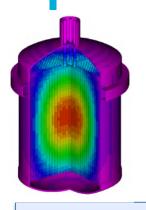


Accidents Modeled

- 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

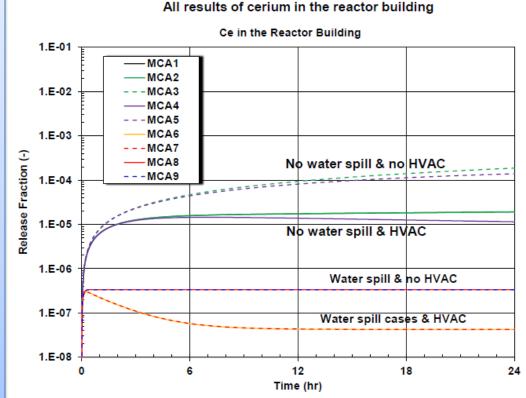


Accidents Modeled

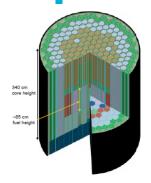
- Full reactor inventory molten salt spill without water
- Full reactor inventory molten salt spill with water

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



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Sodium-Cooled Fast Reactor – ABTR

Power (kW)

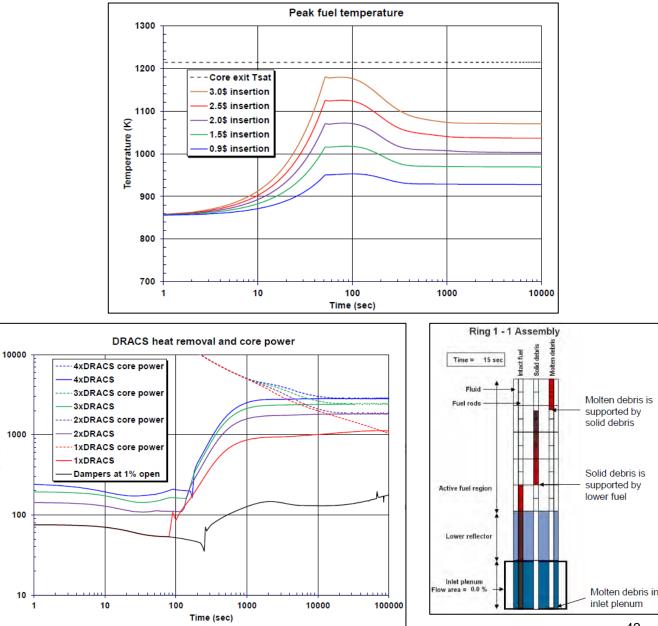


Accidents Modeled

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

Insights Gained on BDBA Behavior

- 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

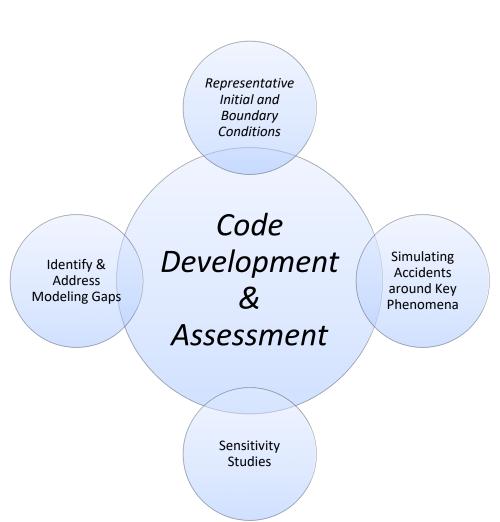
Radionuclide Characterization, Criticality, Shielding, and Transport for the Nuclear Fuel Cycle



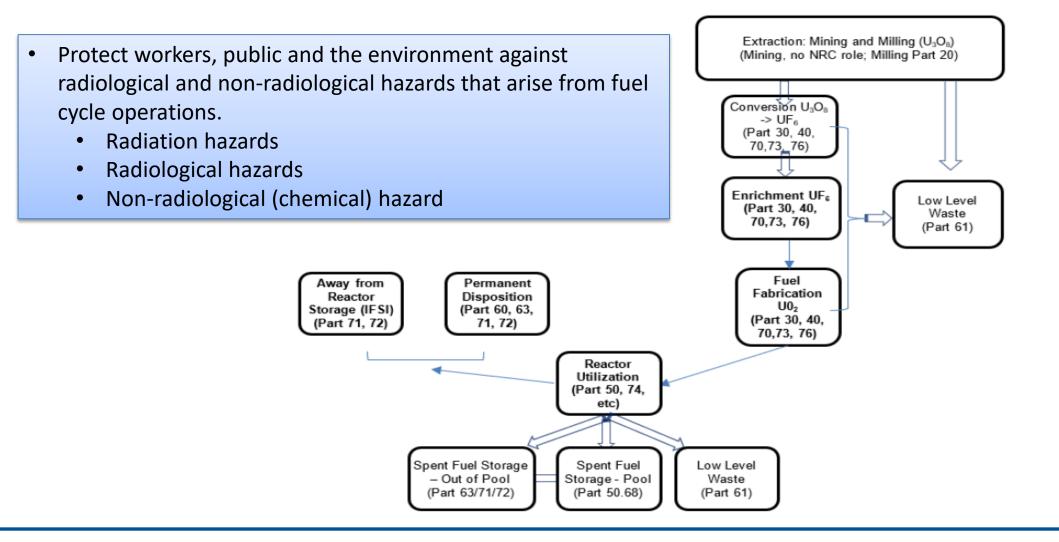
Reminder: General Approach



- 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



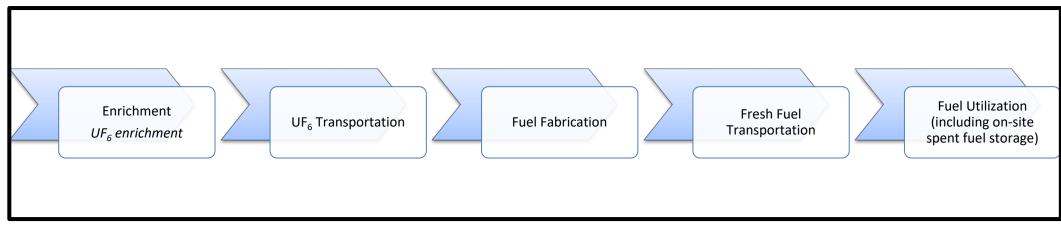
LWR Nuclear Fuel Cycle & Regulations





Project Scope - Non-LWR Fuel Cycle

Stages in scope for Volume 5



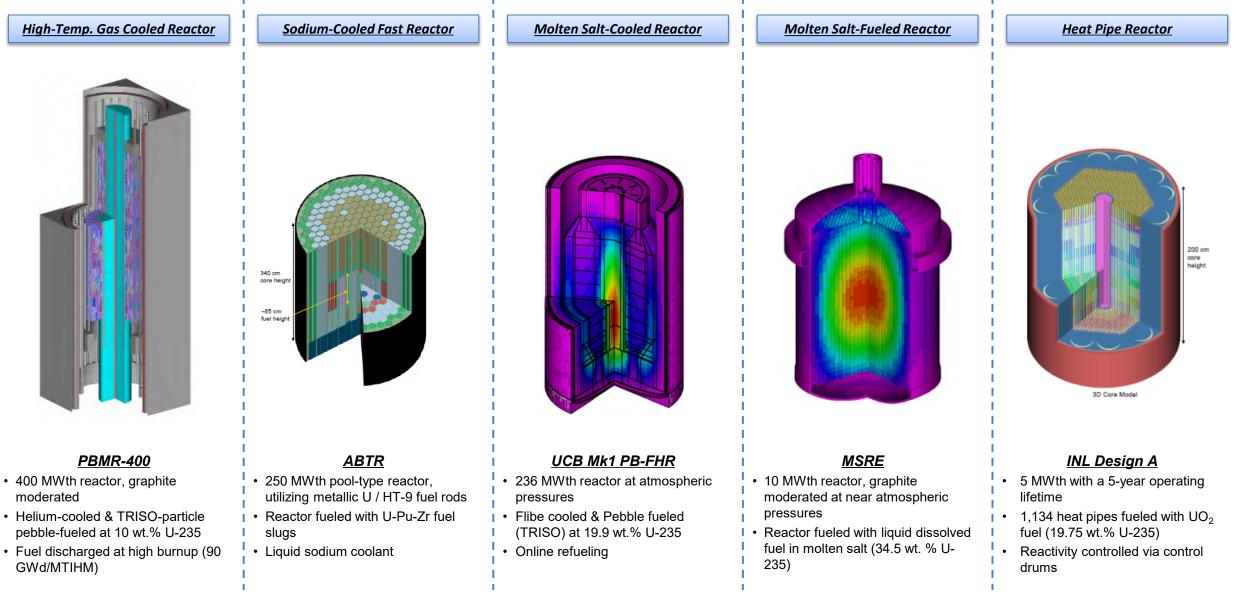
Stages out of scope for Volume 5

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



Non-LWR Designs Considered





Representative Fuel Cycle Designs

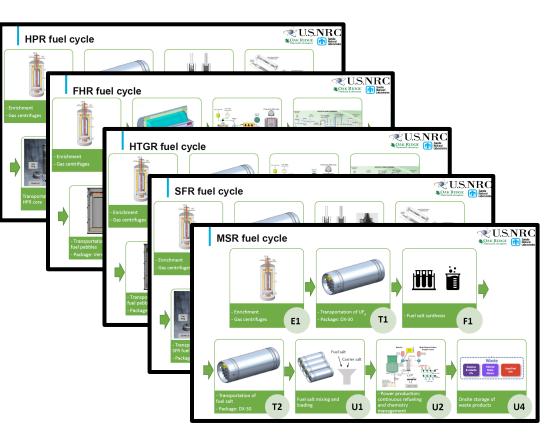
- Developed 5 non-LWR fuel cycle design concepts
 - HPR INL Design A
 - HTGR PBMR-400
 - FHR UCB Mark 1
 - MSR MSRE
 - SFR ABTR

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?

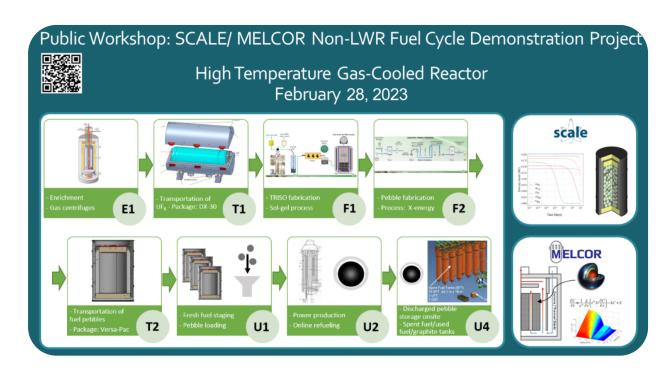
Used as the Initial and Boundary Conditions for developing SCALE & MELCOR models





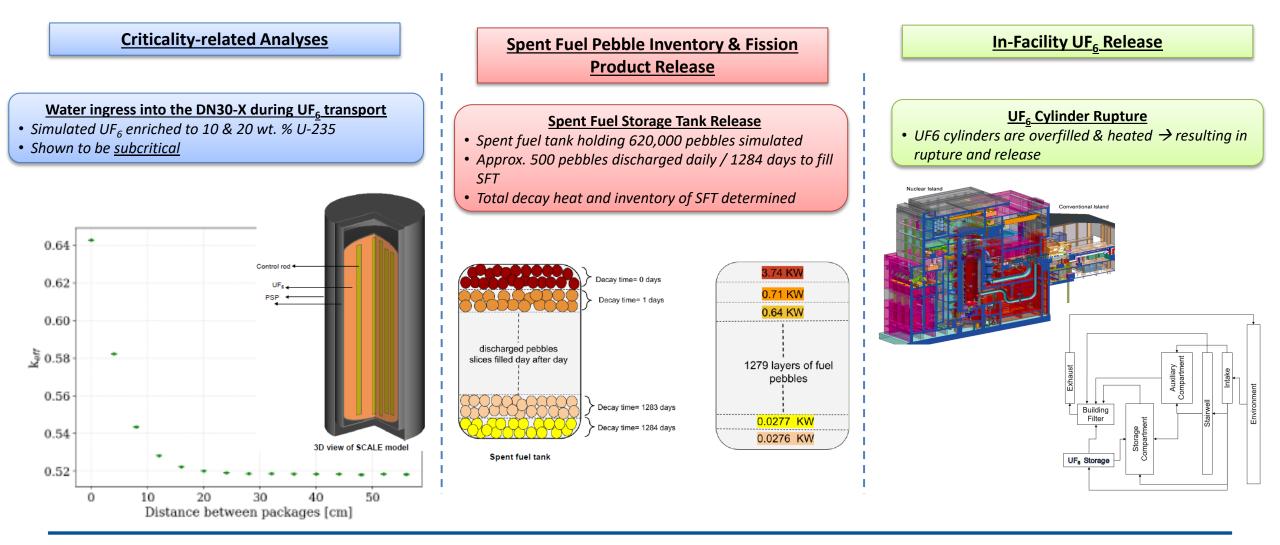
non-LWR Fuel Cycle Demonstration Project –High Temperature Gas-Cooled Reactors

- 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 \rightarrow 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





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





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

Advanced Reactor Stakeholder Public Meeting

Lunch Break

Meeting will resume at 1:00 pm EST

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







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 <u>11 topical reports</u> 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 <u>ACRS letter</u>)
- 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

- <u>April 2022</u>: 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 <u>Matthew.Hiser@nrc.gov</u> 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: <u>https://www.nrc.gov/about-nrc/regulatory/advisory/acrs.html</u>

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 <u>cognizant NRR staff</u> to develop committee engagement plan to optimize review schedule
 - ACRS review completed <u>after</u> 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 <u>https://www.nrc.gov/about-nrc/regulatory/advisory/acrs.html</u>.

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

Advanced Reactor Population-Related Siting Considerations

July 20, 2023





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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



NEI

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 acceptable³."

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



76

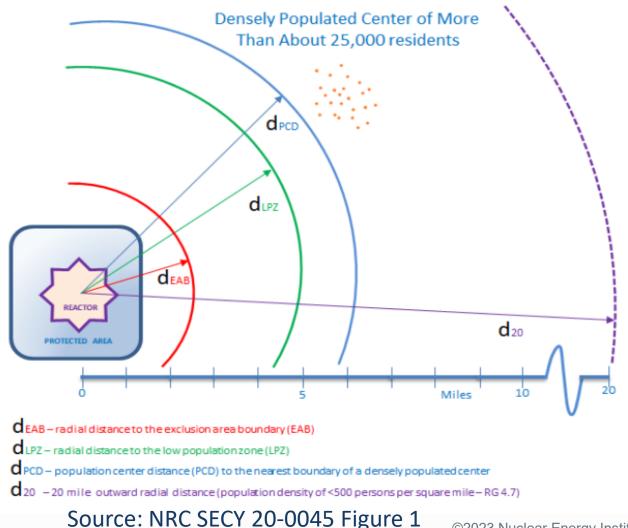
- At least the same degree of protection of the environment, public health & safety, and common defense and security
- Enhanced margins of safety and/or use of simplified, inherent, passive or other innovative means to accomplish safety & security functions
- Designs that incorporate the defense-in-depth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

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temporary appointments.	Responses to Co	Letter No. ADAMS accession No. Commenter		I I I MANUAL I I I I I I I I I I I I I I I I I I I					
	A. General Comm B. Attributes To B	5 ML081900562 North Carolina Waste /	52.47, "Contents of Applications— radio Technical Information," applicants for a released		comprehension, reliable system function, and more straightforward		undertake to review and comment on	SUMMARY: NCUA is amending its share	
(b) * * * (4) Appointment under 5 U.S.C. 3112	Design	tion Network.	design certification must include in • I	reactors to modify their designs to now identify "relationships" between	engineering analysis.	nay promote more	new design concepts, the applicants are responsible for documentation and	insurance rules to simplify coverage for revocable trust accounts. The	
(4) Appointment under 5 U.S.C. 3112 (veterans with compensable service-	C. Security of Adv	6 ML081910787 Nuclear Energy Institute 7 ML081910796 Union of Concerned Sci		include all of the design attributes in the the design attributes and the GDC. No	 Designs that minimize the potential 	e design reviews. of a particular	responsible for documentation and research necessary to support a specific		
connected disability of 30 percent of	D. Relationship to	7	design-specific PRA and its results. In result	Advanced Reactor Policy Statement. changes were made to the policy	for severe accidents and their	cessarily mean that	application. Research activities would	to understand and apply without	
more). The disability must be	(GDC) E. Other Commen	o millionario milliona	addition, 10 CFR 50.71(h) requires each NB	Such changes would essentially result statement as a result of this comment.	consequences by providing sufficient	teria will attach to	include testing of new safety or security		
documented by a notice of retirement or	III. Final Policy Stat		holder of a COL to develop and these	in those plants being completely n out of	inherent safety, reliability, redundancy,	s with some or all	features that differ from existing designs		
discharge due to service-connected	I. Background	This document places each public and time-depe	maintain a PRA for their facility and to curre	redesigned. There is no need for such a	diversity, and independence in safety		for operating reactors, or that use	event of a credit union closing, and help	
disability from active military service		comment into one of the following interactions w		drastic step, given that the NRC Comment: The commenter did not	systems, with an emphasis on	ood by the general	simplified, inherent, passive means to	improve public confidence in the credit	
dated at any time, or by a notice of	On July 8, 1986	categories: indicated that	plant changes and any NRC-endorsed in the consensus standards on PRA. In they	continues to believe that all currently operating reactors provide reasonable to the ARPS, but instead submitted	minimizing the potential for accidents over minimizing the consequences of	umber and nature tirements may	accomplish their safety or security	union system. The amendments	
	Commission publ statement on the r	A. General Comments water reactors'			such accidents.	t to which an	function. The testing shall ensure that these new features will perform as	eliminate the concept of "qualifying beneficiaries." Also, for members with	
Department of Veterans Affairs, dated 1991 or later;	reactors. The Com	B. Attributes To Be Considered that were licen	B provincements the Commission has also fortur	changes were made to the policy (TEN AB-1)	Designs that provide reliable	reactor design	predicted, will provide for the	revocable trust accounts totaling no	
1991 of later,	following three pr	During Design Accordingly, a C. Security of Advanced Reactors to section III."	issued policy statements on the use of DIEVE	statement as a result of this comment. NRC Response: The commenter did	equipment in the balance of plant (BOP)	attributes such as	collection of sufficient data to validate	more than \$500,000, coverage will be	
	issuing the advance		PRA in regulatory activities (60 FR relying	Comment: The commenter states that not address any topic of the draft	(or safety-system independence from	ly.	computer codes, and will show that the	determined without regard to the	
[FR Doc. E8-24227 Filed 10-10-08; 8:45 am]	statement (ARPS):	E. Other Comments Comment: T	42622; August 16, 1995), and severe chan;			mmission expects	effects of system interactions are	proportional beneficial interest of each	
BILLING CODE 6325-39-P	 To maintain t 	Within each category, the NRC has that the discus	accidents regarding future designs and states	secure and that advanced reactors can the comment explain why it should	challenges to safety systems.	es of these	acceptable.	beneficiary in the trust.	
	interaction of app	either repeated comments as written by rulemaking on	existing plants (50 FR 32138; August 8, C. Se	wait until present deficiencies are fixed include design information on a specific	 Designs that provide easily 	igns will be	During the initial phase of advanced	Under the amended rules, a trust	
NUCLEAR REGULATORY	government agenc	the commenter or summarized the Impacts for Ne	1985). The use of PRA as a design tool	and proposed reactors are made safe and secure. (NCWARN-5) design concept. No changes were made to the policy statement as a result of this	maintainable equipment and	e operational cy Planning (EP).	reactor development, the Commission	account owner with up to five different	
COMMISSION	 To provide al including the pub 	comments for conciseness and clarity. (Rulemaking D	the use of PRA and the NRC believes advan	NRC Response: The NRC believes that comment.	 Designs that reduce potential 	orogram, in turn,	particularly encourages design	beneficiaries named in all of his or her	
	Commission's vie	At the end of the comment or comment more akin to b	that the current regulations and policy CODSI	the existing fleet of nuclear power	 Designs that reduce potential radiation exposures to plant personnel. 	d by inspections,	innovations that enhance safety, reliability, and security (such as those	revocable trust accounts at one NCUA- insured institution will be insured up to	
10 CFR Part 50	desired characteri	summary, the NRC references the than a lasting a specific public comments and the letters policy and rec	statements provide sufficient guidance and/	plants is safe and secure. The NRC also III. Final Policy Statement	 Designs that incorporate the 	cceptance criteria	reliability, and security (such as those described previously) and that generally		
[NRC-2008-0237]	reactor designs.	specific public comments and the letters policy and rec by which they were provided to the paragraph or p	to designers. No changes were made to more	believes that advanced reactor designers Consistent with its legislative	defense-in-depth philosophy by	aplementation of	depend on technology that is either	trust account owners with more than	
[nHG-2008-0237]	 To express th 	NRC using the NRC-assigned sequential Background se	the policy statement as a result of this postu-	should consider the expectations in the mandate, the Commission's policy with	maintaining multiple barriers against	5. The Commission	proven or can be demonstrated by a	\$500,000 and more than five different	
Policy Statement on the Regulation of	to issue timely cor	comment numbers listed in Table 1. For NRC Respon	s comment. possi	policy statement to ensure that security respect to regulating nuclear power	radiation release, and by reducing the	anced reactor	straightforward technology development	beneficiaries named in the trust(s) will	
Advanced Reactors	implications of su	example, specific comments are the comment a	n B. Attributes To Be Considered During (NCW	and emergency response are considered reactors is to ensure adequate protection	potential for, and consequences of, severe accidents.	with the	program. In the absence of a significant	be insured for the greater of either:	
AGENCY: Nuclear Regulatory	and the regulatory	referenced as [XXX]-[YYY], where discussion to t	h Design NB	alongside safety during the early stages of the environment and public health of plant design. The fact that such and safety and the common defense and	 Design features that can be proven 	goal policy 144; August 4, 1986,	history of operating experience on an	\$500,000 or the aggregate amount of all	
	On July 12, 199 Commission revis	[XXX] represents the commenter's order to allevia		actions might reduce the need for security. Regarding advanced reactors.	by citation of existing technology, or	iblished at 51 FR	advanced concept reactor, plans for the	the beneficiaries' interests in the	
	addressing the Co	abbreviation and [YYY] represents the statement again	recommends that the policy statement expect	operator actions or improve the overall the Commission expects, as a minimum,	that can be satisfactorily established by	86), and the policy	innovative use of proven technology and/or new technology development	trust(s), limited to \$100,000 per beneficiary.	
	metrication (57 FF	NRC-assigned sequential comment number. Comment: T	explicitly discuss the threat of theft, in design	risk profile for future plants does not at least the same degree of protection of	commitment to a suitable technology	sion to the metric	programs should be presented to the		
SUMMARY: On May 9, 2008 (73 FR	1992; as revised Ju	the NBC limes	h addition to the current focus on threat large.	mean that the existing operating plants the environment and public health and	development program.	(61 FR 31169; June	NRC for review as early as possible, so	DATES: This rule is effective on October 14, 2008, Written comments must be	
26349), the Nuclear Regulatory	31169)).	A. General Comments difficult hurdle	of sabotage of facilities, and encourage curre designers to consider requirements for rulen	are unsafe. No changes were made to the safety and the common defense and	 Designs that include considerations 		that the NRC can assess how the	14, 2008. Written comments must be received on or before December 15.	
Commission (NRC; Commission) issued,		Comment: The commenter believes and wants the	designers to consider requirements for ruled	policy statement as a result of this security that is required for current	for safety and security requirements	e timely and d advanced	proposed program might influence	received on or before December 15, 2008.	
	2001, the NRC has	that a fast fuel reactor can help reduce a pilot program	monitoring early in the design process. Desig	comment. generation light-water reactors (LWRs). ¹ Furthermore, the Commission expects	together in the design process such that security issues (e.g., newly identified	sion encourages	regulatory requirements.	ADDRESSES: You may submit comments	
	threats and their p Nation's fleet of o	the volume of radioactive waste bureaus would			threats of terrorist attacks) can be	interaction of	Finally, the NRC also believes that it	ADDRESSES: You may submit comments by any of the following methods (Please	
reinforces the Commission's current	power reactors an	currently in storage at reactor sites in review, license the United States and hopes that the designs. (Walk	located with representing facilities. The that r	Criteria (GDC) that advanced reactors will provide enhanced margins of safety and/or use	effectively resolved through facility	other government	will be in the interest of the public as	send comments by one method only):	
policy regarding advanced reactors and	upgrades of physi	the United States and hopes that the NRC has considered or will consider NRC Respon	e commenter suggested a possible aircra	Comment: The commenter wants the simplified, inherent, passive, or other	design and engineered security features.	C to provide for	well as the design vendors and the	 Federal eRulemaking Portal: http:// 	
includes new items to be considered	and mitigative str	those designs. (Sund-1) Act of 1974, as	addition to the list of design attributes from	agency to incorporate the 'expectations' innovative means to accomplish their	and formulation of mitigation measures,	f regulatory	prospective license applicants to address security issues early in the	www.regulations.gov. Follow the	
during the design of these reactors,	issuance of a serie	NRC Response: The NRC neither NRC's response	al included in the policy statement that robus	in the policy statement into the safety and security functions.	with reduced reliance on human	anced reactors and	design stage to achieve a more robust	instructions for submitting comments.	
including security, emergency	and license condi	develops nor promotes reactor designs, responsibilitie	relates to theft and international Howe	regulations as additional GDC. (UCS-1) Among the attributes that could assist	actions.	ted parties,	and effective security posture for future	 NCUA Web Site: http:// 	
preparedness, threat of theft, and	power reactors, th	but rather reviews the safety and nuclear reactor		NRC Response: The GDC establish in establishing the acceptability or	 Designs with features to prevent a simultaneous loss of containment 	with a timely, sent of the safety	nuclear power reactors.	www.ncua.gov/ RegulationsOpinionsLaws/	
international safeguards.	considers it prude	security aspects of designs proposed by cannot transfer	the comment and has added need	minimum requirements for the principal design criteria for nuclear power plants.	simultaneous loss of containment integrity (including situations where the	ent of the safety ristics of advanced	Dated at Rockville, Maryland, this 7th day	proposed regs/proposed regs.html.	
		reactor vendors and designers. The NRC another entity.	4 second statement and second seco	design criteria for nuclear power plants. The goal of the policy statement is not	containment is by-passed), and the	a licensing	of October 2008.	Follow the instructions for submitting	
2008.	matters to prospec they can use this i	has the ability to develop the capability to the policy st	consider the threat of theft and with	to raise these minimum requirements,	ability to maintain core cooling as a	ince early in the	For the Nuclear Regulatory Commission.	comments.	
FOR FURTHER INFORMATION CONTACT: Mr.	the design stage of	to evaluate innovative and advanced comment. designs that are presented for NRC Comment: T	requirements for implementing all lic	but rather to encourage advanced shutdown and decay heat removal	result of an aircraft impact, or	ontribute towards	Annette L. Vietti-Cook,	 E-mail: Address to 	
Wesley W. Held, Rulemaking, Guidance, and Advanced Reactor Branch, Division	identify notential	review (e.g., Toshiba's 4S reactor "type-licensing		reactor designers to consider safety and systems. The use of inherent or nassive	identification of system designs that	ity and adding	Secretary of the Commission.	regcomments@ncua.gov. Include "[Your	
of New Reactor Licensing, Office of New	and/or design feat	design). No changes were made to the "fast-track con	early in the design phase. An attribute to de	security matters during the development means to accomplish this objective is	would provide inherent delay in	ability in the	[FR Doc. E8-24268 Filed 10-10-08; 8:45 am]	name] Comments on Share Insurance for	
Reactors, U.S. Nuclear Regulatory	more robust and e	policy statement as a result of this licenses " (Wa	has been added to the list of design (DB1	of future reactor designs. No changes encouraged (negative temperature	radiological releases (if prevention of	ion of advanced	BILLING CODE 7590-01-P	Revocable Trust Accounts" in the e-mail	
Commission, Mail Stop: T-6 C34,	posture. Therefore	comment. NRC Respon	s attributes to be considered during the No ct	were made to the policy statement as a coefficient, natural circulation, etc.).	release is not possible).	s not develop new		subject line.	
Washington, DC 20555-0001;	decided to revise t	Comment: The commenter suggests agrees with the		result of this comment. Comment: The commenter believes • Longer time constants and sufficient instrumentation to allow for	 Designs with features to prevent loss of spent fuel pool integrity as a 	commission intends		 Fax: (703) 518–6319. Use the subject line described above for e-mail. 	
Telephone: (301) 415-1583; fax number:	these expectations	that the term "current generation" in the commenter no		that the utility of the policy could be more diagnosis and management before	result of an aircraft impact.	ility, when	NATIONAL CREDIT UNION	 Mail: Address to Mary Rupp, 	
(301) 415-5399; e-mail:	emergency prepar	first paragraph of the policy statement regulations in	that the following attributes in the the A	enhanced if the relationship of the reaching safety systems challenge and/	 Designs with features to eliminate 	ly assessment and	ADMINISTRATION	Secretary of the Board, National Credit	
Wesley.Held@nrc.gov.	current expectatio The Commissio	could be misinterpreted because it was written in 1986 and does not take into CFR Part 52, "	current ARPS are not statements of react	attributes listed in the policy to the GDC or exposure of vital equipment to	or reduce the potential theft of nuclear	re and advanced		Union Administration, 1775 Duke	
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		suggests that the term "current certification" t	should be deleted. aviati	NRC Response: The NRC believes that where possible, reduce required	barriers to potential theft of nuclear	tor designs—even	HIN 3133-AU54	 Hand Delivery/Courier: Same as 	
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2		NRC Response: The NRC agrees that COL is a licens		that safety and security are systems should facilitate operator	NRC for comment and/or evaluation, the	is are identified, the	AGENCY: National Credit Union Administration (NCUA).	RegulationsOpinionsLaws/comments as	
	um ElectroNuclear AB	the term "current generation" may in the granting cause confusion because it is subjective construction p	result of an aircraft impact, or attack	appropriately considered in the design.	Commission can develop preliminary	solution can be	ACTION: Interim final rule with request	submitted, except as may not be	©2023 Nuclear E
		cause confusion because it is subjective construction p	e identification of system designs that secur	The attributes do not necessarily	design safety evaluation and licensing	e applicants are	ACTION: Interim final rule with request for comments.	possible for technical reasons. Public	SZUZJ NUUICAI L
	I		would provide inherent delay in exerc	correspond to any particular GDC or set power plants licensed before 1997.	criteria for their safety-related and	the NRC will	ior commettus.	comments will not be edited to remove	
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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)





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 d₃₀
- - 500 ppsm, different than what was proposed in WASH-1308
 - No reference to a dose basis
- RG 4.7, R2 (1998) changed PDD to d₂₀
 - 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, designspecific 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 wellunderstood
 - 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?



- 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

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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)
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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 manmade 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.

Advanced Reactor Stakeholder Public Meeting

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 Reactor Systems Engineer Accident Analysis Branch, Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission



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



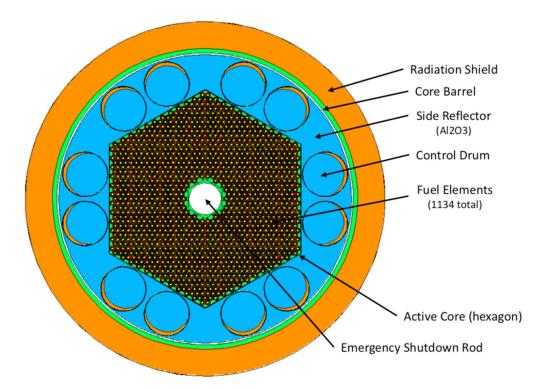
Project Approach

- 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, 5th quantile, and 95th 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.

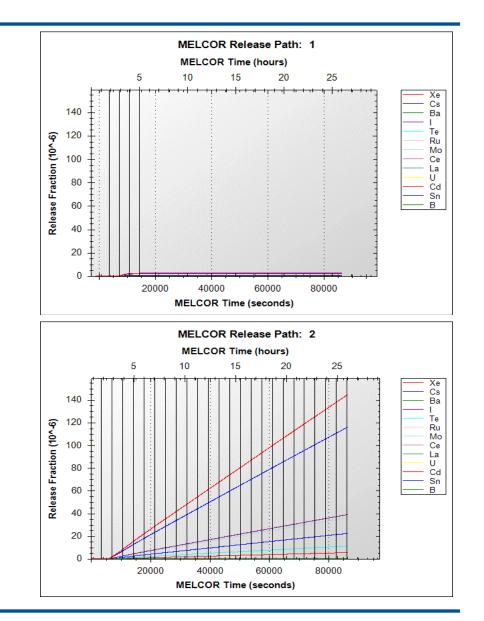


INL Design A reactor vessel cross-section (from figure 3-1 of SAND2022-2745)



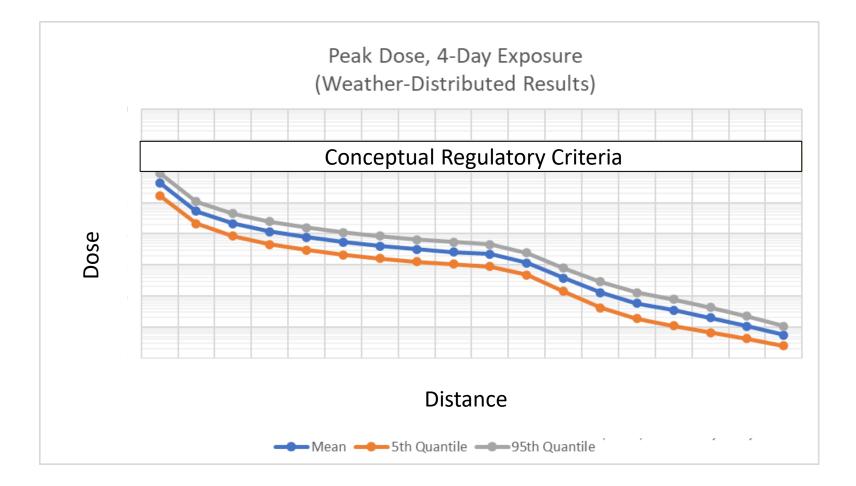
Source Term*

- 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.





Example of Main Analysis Results*





Sensitivity Analysis of Radionuclides

• MACCS guidance recommends using a subset of radionuclides from the LWR core for analysis

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	

 That list was expanded to include additional radionuclides from the conceptual HPR core

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

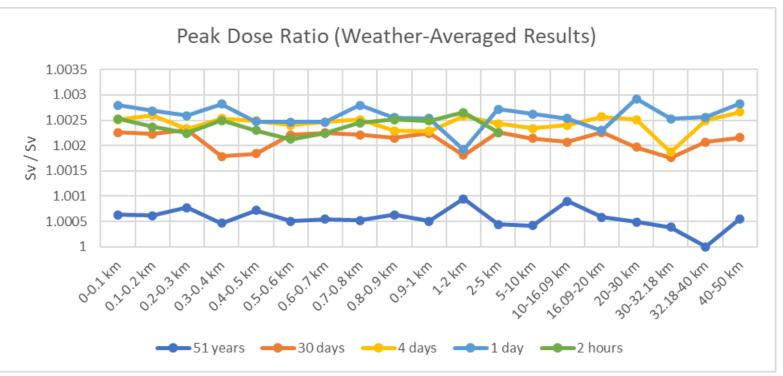
Additional Daughter Radionuclides

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

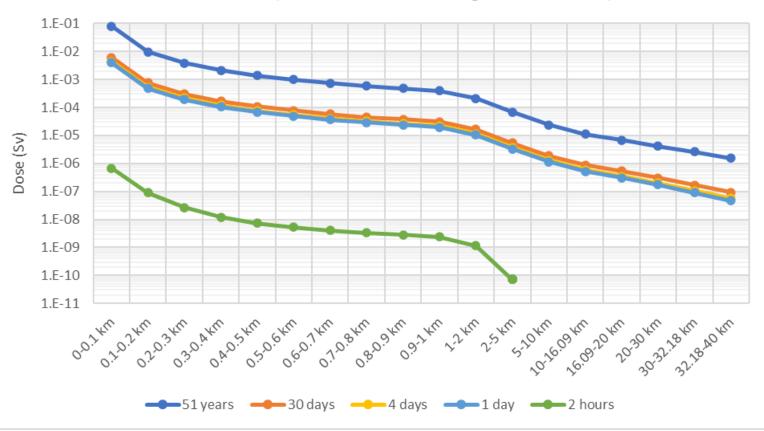




Sensitivity Analysis of Dose Exposure Period*

- 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).

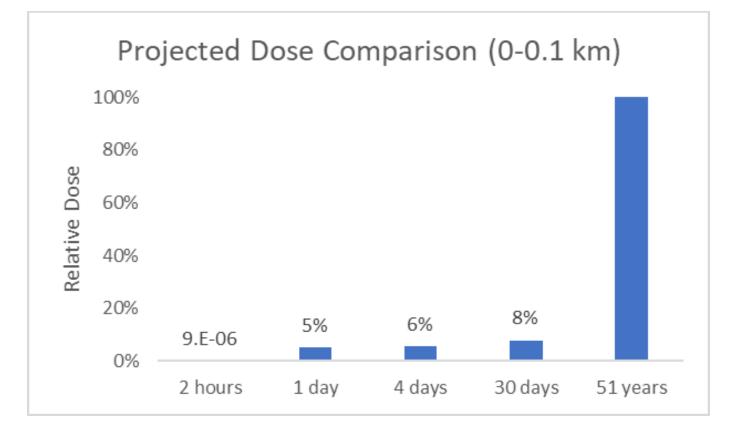
Peak Dose (Weather-Averaged Results)





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





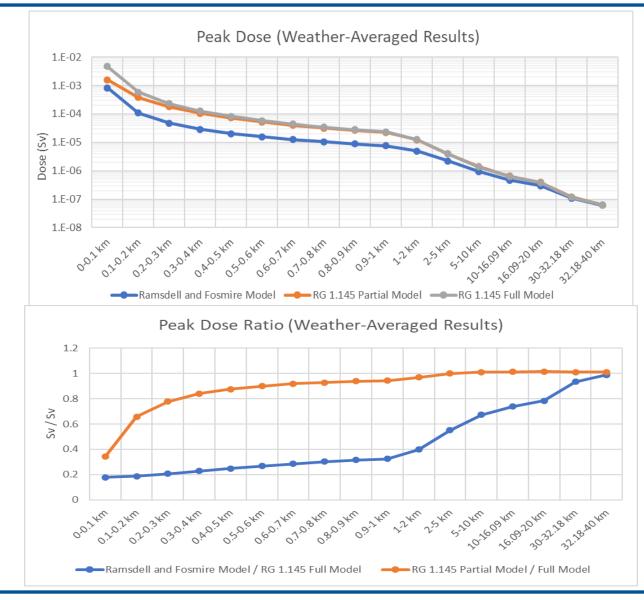
Sensitivity Analysis of Nearfield Modeling

- 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*

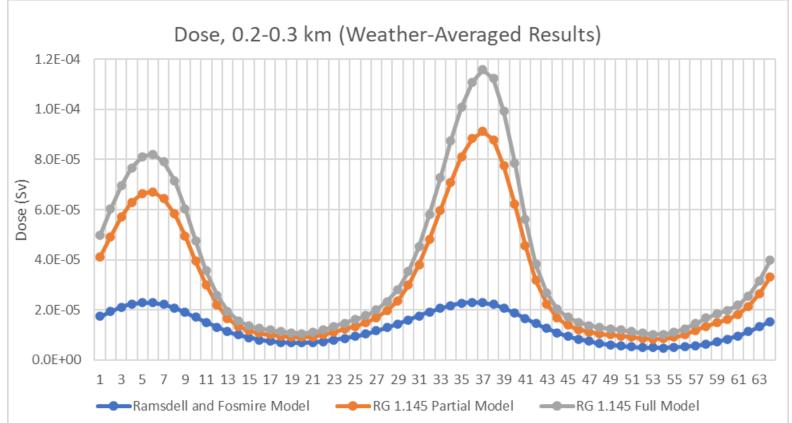
- The amount of plume spread in the different models notably impacts doses in the first 40 km (25 mi).
- Because the Ramsdell and Fosmire Model has the most spread, it has the lowest peak doses of the three nearfield models.
- The two RG 1.145 models have similar peak doses after roughly 1 km (0.62 mi),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





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.





Sensitivity Analysis of Release Timing

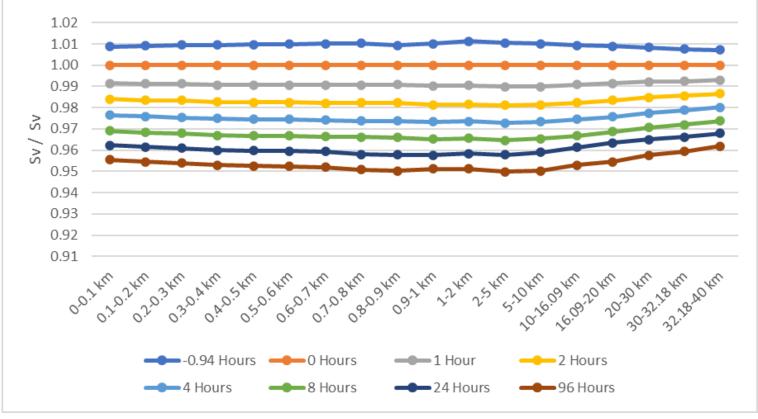
- MACCS assumes that reactor shutdown occurs at the beginning of an accident scenario. MACCS begins calculating decay and ingrowth at this time.
- The TOP scenario is different in that shutdown does not immediately occur.
- The reactor is postulated to operate for roughly an hour when the reactor power 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).

Peak Dose Ratio (Weather-Averaged Results)



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. <u>ML23045A044</u>
- E. Walker, S.E. Skutnik, W.A. Wieselquist, A. Shaw, and F. Bostelmann, "SCALE Modeling of the Fast-Spectrum Heat Pipe Reactor," ORNL/TM-2021/2021, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 2021, <u>ML22158A054</u>.
- 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, <u>ML22144A188</u>.
- 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, <u>ML22270A046</u>.
- Sandia National Laboratories, "Technical Bases for Consequence Analysis Using MACCS (MELCOR Accident Consequence Code System)," NUREG/CR-7270, U.S. Nuclear Regulatory Commission, Washington, DC, October 2022, <u>ML22294A091</u>.
- D.J. Clayton, "Implementation of Additional Models into the MACCS Code for Nearfield Consequence Analysis," SAND2021-6924, Sandia National Laboratories, Albuquerque, New Mexico, June 2021, <u>ML21257A120</u>.



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



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

Factory or Manufacturing Facility

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Fabricate the reactor, load fuel, and potentially operate the reactor for functional testing Transportation to the Deployment Site



Reactors may contain fresh or irradiated fuel



Deployment Site – Power Operation

Stand-alone, self-contained micro-reactor design

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Core module with onsite reactor building and power conversion equipment Transportation from the Deployment Site

Reactors may contain spent or irradiated fuel

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Decommissioning or Refurbishing Facility

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Remove fuel and decommission the reactor, recycle components and systems, or refurbish and refuel the reactor for redeployment

Deployment Lifecycle



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



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



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



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



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 riskinformed, 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



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



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



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



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



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



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



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.



Advanced Reactor Stakeholder Public Meeting

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



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

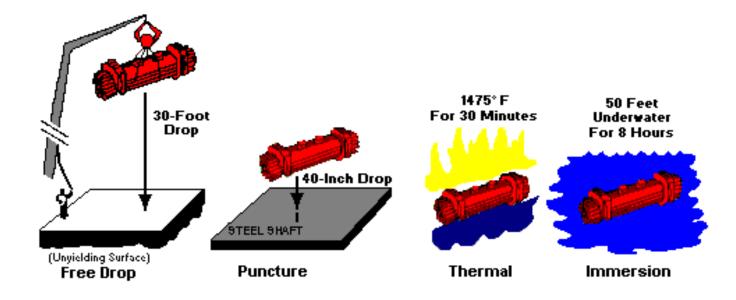


Package Approval Standards

- 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 (<u>ML23066A201</u>) 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





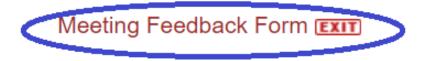
How Did We Do?

• 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 Dates and Times



