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

Advanced Reactor Stakeholder Public Meeting

January 19, 2022

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 323 588 045#

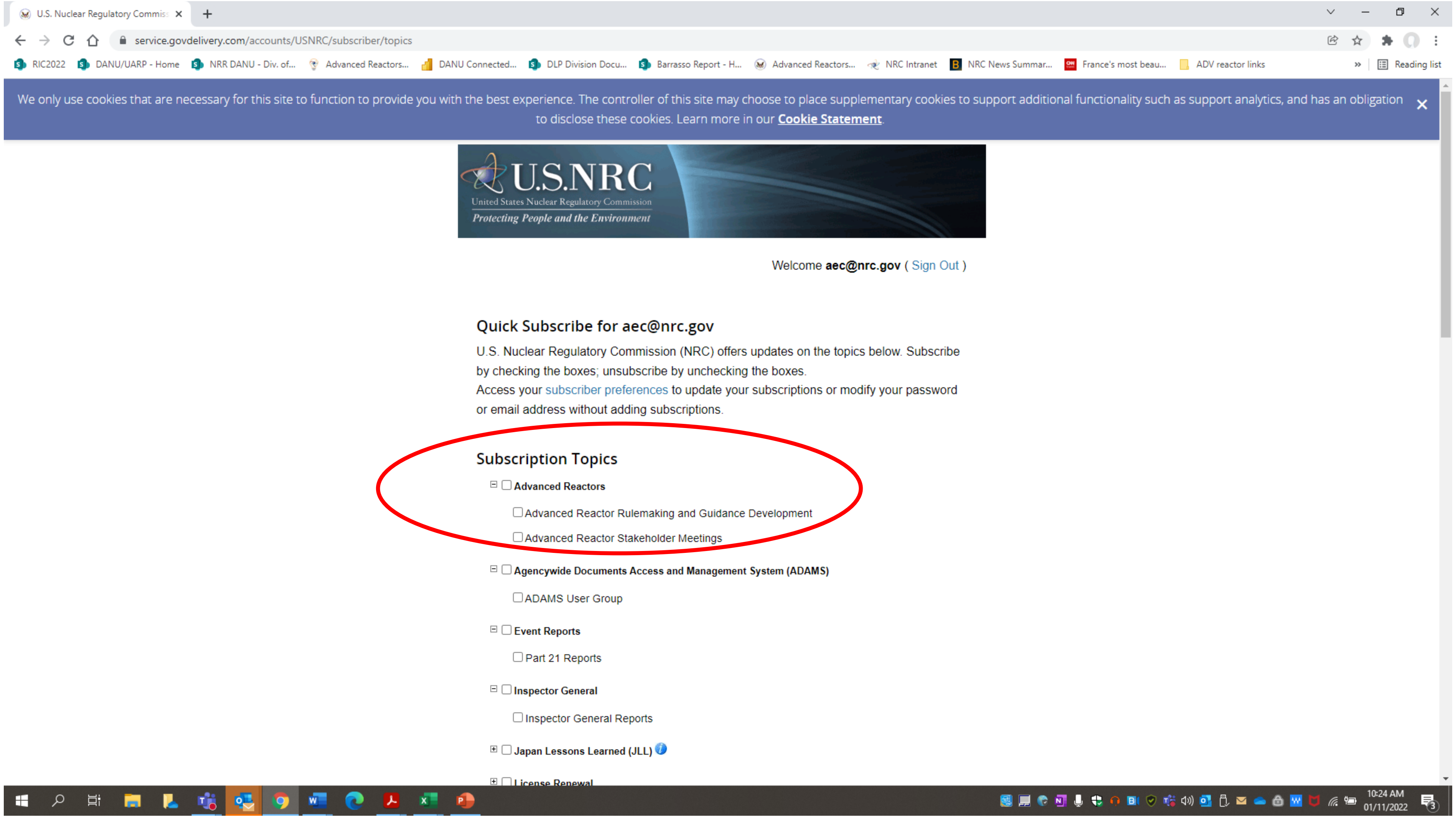
Advanced Reactor Program

New names of GovDelivery categories: from "NRC-DOE non-LWR workshops" to "Advanced Reactor Stakeholder Meetings"; from "Advanced Reactor Guidance Initiative" to "Advanced Reactor Rulemaking and Guidance Development"

https://service.govdelivery.com/accounts/USNRC/subscriber/new

Slide 3

https://service.govdelivery.com/accounts/USNRC/subscriber/topics



Time	Agenda	Speaker
10:00 – 10:20 am	Opening Remarks / Adv. Rx Integrated Schedule	NRC
10:20 – 10:30 am	Status Overview of the Adv. Rx Generic Environmental Impact Statement (GEIS) and Rulemaking Activities	NRC
10:30 – 11:15 am	Implementing Near-field Models in MACCS v4.1 for Better Near-field Dose Calculations	NRC/SNL
11:15 am – 12:00 pm	Light Water Reactor Construction Permit Interim Staff Guidance	NRC
12:00 – 1:00 pm	Lunch Break	All
1:00 – 1:45 pm	Nuclear Data Assessment for Advanced Reactors	NRC/ORNL
1:45 – 2:30 pm	SCALE/MELCOR Development and Applications for non-LWRs	NRC/SNL & ORNL
2:30 – 2:40 pm	Break	All
2:40 – 3:20 pm	Advanced Manufacturing Technologies	NRC
3:20 – 3:30 pm	Future Meeting Planning and Concluding Remarks	NRC

Advanced Reactor Integrated Schedule of Activities

The updated Advanced Reactor Integrated Schedule

is publicly available on NRC Advanced Reactors website at:

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

Advanced Reactor Integrated Schedule of Activities

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

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

Legend

Concurrence (Division/Interoffice)	• EDO Concurrence Period
Federal Register Publication	Commission Review Period**
Public Comment Period	▼ ACRS SC/FC (Scheduled or Planned)
Draft Issuance of Deliverable	External Stakeholder Interactions
Final Issuance of Deliverable	↓ Public Meeting (Scheduled or Planned)

Version 1/7/22

Present Day

Strategy	Regulatory Activity	Commission Papers	Guidance	Rulemaking	NEIMA	Complete	2021 Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	2022 Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
1	Development of non-Light Water Reactor (LWR) Training for Advanced Reactors (Adv. Rxs) (NEIMA Section 103(a)(5))					x																								
	FAST Reactor Technology				x	x																								
	High Temperature Gas-cooled Reactor (HTGR) Technology				x	x																								
	Molten Salt Reactor (MSR) Technology				x	x																								
	Competency Modeling to ensure adequate workforce skillset					x																								
2	Identification and Assessment of Available Codes					x																								
	Development of Non-LWR Computer Models and Analytical Tools																													
	Code Assessment Reports Volume 1 (Systems Analysis)					x																								
	Reference plant model for Heat Pipe-Cooled Micro Reactor (update from v1 to v2)					v1																					v2				
	Reference plant model for Sodium-Cooled Fast Reactor (update from v1 to v2)					v1																						v2			
	Reference plant model for Molten-Salt-Cooled Pebble Bed Reactor					x																								
	Reference plant model for Monolith-type Micro-Reactor																													
	Reference plant model for Gas-Cooled Pebble Bed Reactor																													
	Code Assessment Reports Volume 2 (Fuel Perf. Anaylsis)					x																								
	FAST code assessment for metallic fuel					x																								
	FAST code assessment for TRISO fuel					x																								
	Code Assessment Reports Volume 3 (Source Term Analysis)					x																								
	Non-LWR MELCOR (Source Term) Demonstration Project					x						↓	↓		↓															
	Reference SCALE/MELCOR plant model for Heat Pipe-Cooled Micro Reactor					x																								
	Reference SCALE/MELCOR plant model for High-Temperature Gas-Cooled Reactor					x																								
	Reference SCALE/MELCOR plant model for Molten Salt Cooled Pebble Bed Reactor					x																								
	Reference SCALE/MELCOR plant model for Sodium-Cooled Fast Reactor																													
	Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor																													
	MACCS radionuclide screening analysis																													
	MACCS near-field atmospheric transport and dispersion model assessment					x																								
	MACCS radionuclide properties on atmospheric transport and dosimetry																													
	MACCS near-field atmospheric transport and dispersion model improvement					x																								
	Code Assessment Report Volume 4 (Licensing and Siting Dose Assessments)							▼																						
	Phase 1 – Atmospheric Code Consolidation																													
	Code Assessment Report Volume 5 (Fuel Cycle Analysis)					x		▼																						
	Research plan and accomplishments in Materials, Chemistry, and Component Integrity for Adv. Rxs.					x																								
	Research on risk-informed and performance-based (RIPB) seismic design approaches and adopting seismic isolation technologies																													

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

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

Advanced Reactor Integrated Schedule of Activities

UPDATES:

Strategy 2, "Computer Codes and Review Tools":

- "Reference plant model for Heat Pipe-Cooled Micro Reactor" – task complete
- "Reference plant model for Sodium-Cooled Fast Reactor (update from version 1 to 2)" – v1 complete; v2 completion Sept. 2022
- "Reference plant model for Monolith-type Micro-Reactor" – completion Jul. 2022
- "Reference plant model for Gas-Cooled Pebble Bed Reactor" – completion Dec. 2022
- "MACCS near-field atmospheric transport and dispersion model assessment" – Marked complete
- "MACCS radionuclide properties on atmospheric transport and dosimetry" – Final issuance of deliverable now Sept. 2022 from June 2022

Strategy 3, "Guidance":

- "Develop Advanced Reactor Technology Inclusive Content of Application Project (TICAP) Regulatory Guidance" - Added a TICAP public meeting in January 2022
- "Develop Advanced Reactor Inspection and Oversight Framework Document" – Draft issuance of deliverable moved to February 2022 from December 2021
- "Develop Environmental ISG for Micro Reactors" – item complete and no longer being tracked – removed

Advanced Reactor Integrated Schedule of Activities

UPDATES (contd.):

Strategy 3, "Guidance" (contd.):

- "Develop MC&A guidance for Cat II facilities (NUREG-2159)" - Draft of NUREG at end of Sept. 2021; 60-day comment period, extended to Dec. 3 per NEI request. Issue final by March 2022 (shifted by five months)

Strategy 4, "Consensus Codes and Standards":

- "Develop Regulatory Guide for endorsement of the ASME Section XI, Division 2 Standard (Reliability and Integrity Management)" - Draft Guide issued 9/30/21; public comment period closed 11/15/21 - staff working to resolve comments; plan to issue Final RG ~June 2022

Strategy 5, "Policy and Key Technical Issues":

- "Report regarding review of the insurance and liability for advanced reactors (Price-Anderson Act)" – completed 12/21/21 (due date 12/31/21)
- "Develop SECY Paper regarding Population-Related Siting Considerations for Advanced Reactors" - marked complete with issuance of SECY-20-0045
- New item: "Revise Regulatory Guide (RG) 4.7 to implement SRM-SECY-20-0045" (SRM not issued yet)

Advanced Reactor Integrated Schedule of Activities

UPDATES (contd.):

Rulemaking:

- “Part 53 Plan - Risk-Informed, Technology Inclusive Regulatory Framework for Advanced Reactors (NEIMA Section 103(a)(4))” – Extension request approved. This version reflects new schedule including interactions with ACRS - concurrence in Sept – Dec 2022; ACRS meetings in Feb, Apr, Jun, Aug-Oct
- “Physical Security for Advanced Reactors” – Extension request approved. Changes reflect new schedule
- “Develop draft Generic Environmental Impact Statement for Advanced Reactors. Final GEIS.*(Has been voted to rulemaking by Comm.)” – Draft issuance of deliverable May 2022
- “Emergency Preparedness Requirements for Small Modular Reactors and Other New Technologies.(NEIMA Section 103(a)(2))” - OEDO concurred and sent the package (SECY-22-0001) to the Commission on December 30, which is now with the Commission for their review and approval

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Advanced Reactor Generic Environmental Impact Statement and Rulemaking Status

Laura Willingham, Environmental Project Manager
Environmental Center of Expertise, U.S. NRC

Rulemaking Process

- The Proposed Rule Package is publicly available while it is with the Commission for review.
  - No public comments taken during the Commission review
  - Commission will vote on publishing the proposed rule package
  - If Commission votes to approve publication of the proposed rule package
    - Proposed rule to be issued in the *Federal Register* with a 75-day public comment period.
    - Public meetings will be held during the comment period
- Advanced Reactor GEIS Rulemaking Website
  - https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/ruledetails.html?id=1139

Current Status & Rulemaking Schedule

November 2021	• Proposed rule submitted to Commission on November 30, 2021.
May 2022 (estimated)	• Proposed rule published for 75-day comment period (if approved by Commission)
May 2023 (estimated)	• Final rule submitted to Commission
Jan 2024 (estimated)	• Final rule publication (if approved by Commission)

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Proposed Rule Package

- Proposed Rule Package can be found using the Accession No. in the Agencywide Document Management System (ADAMS) at https://www.nrc.gov/reading-rm/adams.html#web-based-adams

Document	ADAMS Accession No.
Proposed Rule Package: SECY-21-0098: Proposed Rule: Advanced Nuclear Reactor Generic Environmental Impact Statement (RIB3150-AK55; NRC-2020-0101)	**ML21222A044**
Preliminary Draft Guide-4032 Package: Preliminary Draft Guide-4032 (RG 4.2), "Preparation of Environmental Reports for Nuclear Power Stations"	**ML21208A111**
Preliminary Draft of Interim Staff Guidance COL-ISG-30: Draft Interim Staff Guidance COL-ISG-30: Advanced Reactor Applications – Environmental Considerations for Advanced Nuclear Applications that Reference the Generic Environmental Impact Statement (NUREG-2249)	**ML21227A005**

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Proposed Rule Package (con't)

- Portions of Proposed Package can also be found at Regulations.gov under "Browse Documents" tab at https://www.regulations.gov/docket/NRC-2020-0101/document.
  - The following documents can be found at Regulations.gov
    - SECY paper
    - Draft Advanced Reactor GEIS
    - Draft Guide-4032
    - Draft Regulatory Analysis
    - Draft COL-ISG-30
  - The Docket ID on Regulations.gov for the ANR GEIS is NRC-2020-0101.
  - Hit "Subscribe" to get notifications when new content is added.

QUESTIONS?

Implementing Nearfield Models in MACCS v4.1 for Better Nearfield Dose Calculations

PRESENTED BY

Dan Clayton
MACCS Principal Investigator
Sandia National Laboratories

Advanced Reactor Stakeholder Meeting
January 19, 2022

SAND2022-0282 PE

Agenda

Motivation and Purpose

Background

Approach

- Nearfield Code Comparisons
- MACCS 4.1 Enhancements and Algorithms
- Verification and Comparison

Summary

Motivation and Purpose

Motivation: Resolve the technical issues with the adequacy of MACCS in the nearfield (i.e., at distances less than 500 m) that are identified in a **non-Light Water Reactor (LWR) vision and strategy report** that discusses computer code readiness for non-LWR applications developed by the Nuclear Regulatory Commission (NRC)

The **purpose** of this presentation is threefold:

- **Summarize** the technical issues associated with the use of MACCS in the nearfield and approach used to resolve them
- **Alert** stakeholders that improved nearfield modeling capabilities have been added to MACCS 4.1
- **Familiarize** stakeholders with the improved nearfield capabilities available in MACCS 4.1

Background

MACCS 4.0 uses the general **gaussian plume equation** with reflective boundaries and includes **models** for **plume meander** and **building wake effects** based on building dimensions

$$C = \frac{\dot{Q}}{2\pi\sigma_y\sigma_z u}\exp\left(\frac{-y^2}{2\sigma_y^2}\right)\sum_{n=-\infty}^{\infty}\left\{\exp\left[-\frac{1}{2}\left(\frac{2nh-H-z}{\sigma_z}\right)^2\right]+\exp\left[-\frac{1}{2}\left(\frac{2nh+H-z}{\sigma_z}\right)^2\right]\right\}$$

Previous (4.0 and earlier) versions of MACCS include only a **simple model** for building wake effects. The MACCS User's Guide suggests that this simple building wake model **should not be used at distances closer than 500 m**. This statement raised the question of **whether MACCS can reliably be used to assess nearfield doses**, i.e., at distances less than 500 m

Approach

Identify candidate **codes** considered **adequate** for use in nearfield modeling

Benchmark MACCS 4.0 nearfield results against results from candidate codes

Identify model **input** recommendations or **code updates** for improved nearfield modeling

Implement the code **updates** in MACCS 4.1

Verify that the **MACCS** 4.1 code **updates** adequately reflect the results from the candidate codes

Exercise new capabilities in **MACCS** 4.1

Nearfield Code List

Four **candidate codes** were selected from the three **main methods** of atmospheric transport and dispersion (ATD) in the nearfield and evaluated

- CFD models – OpenFOAM
- Simplified wind-field models – QUIC
- Modified Gaussian models – AERMOD and ARCON96

Based on these rankings, **QUIC**, **AERMOD**, and **ARCON96** were selected for **comparison with MACCS** 4.0 (3.11.6)

Test cases developed varying

- Weather conditions
- Building configurations (HxWxL)
- Power levels (heat content)

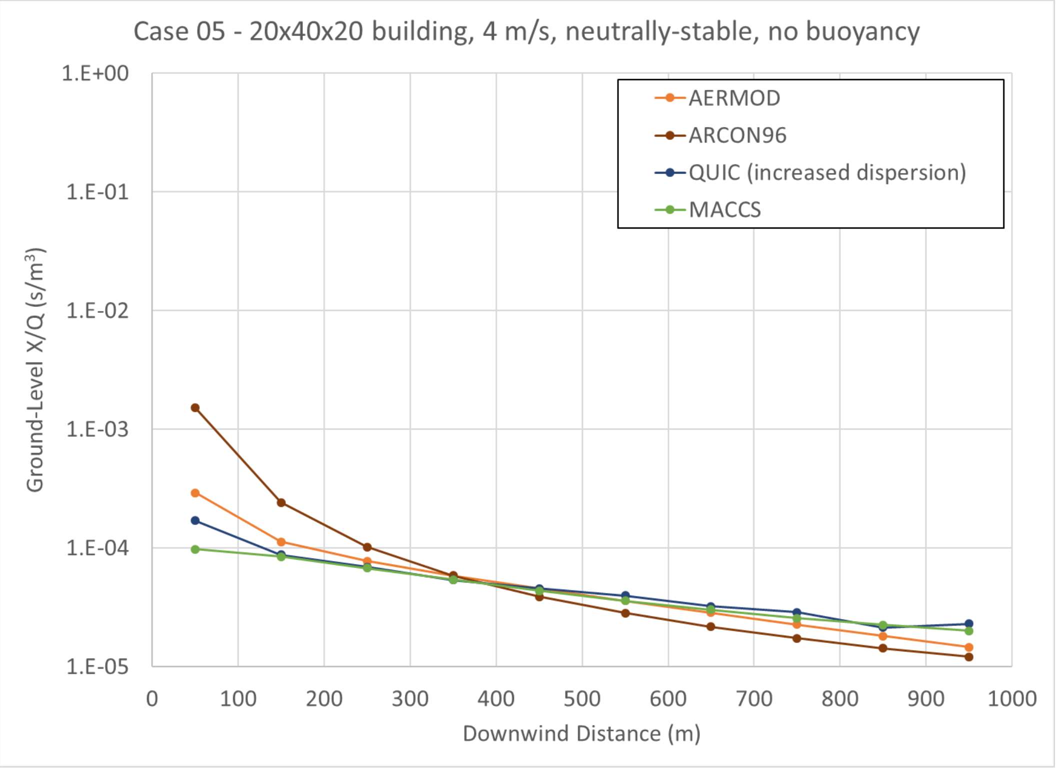
	Model Characteristics					
Model	Simplicity	Efficiency	Validation	Conservative Bias	Community Acceptance	Ease of Implementation
OpenFOAM	3	3	1	2	1	3
QUIC	3	2	1	2	2	3
ARCON96	1	1	2	2	1	1
AERMOD	1	1	1	2	1	2

MACCS 4.0 Nearfield Comparison Results

At 50 m, order from **highest to lowest** is ARCON96, AERMOD, QUIC, MACCS

Order changes with distance

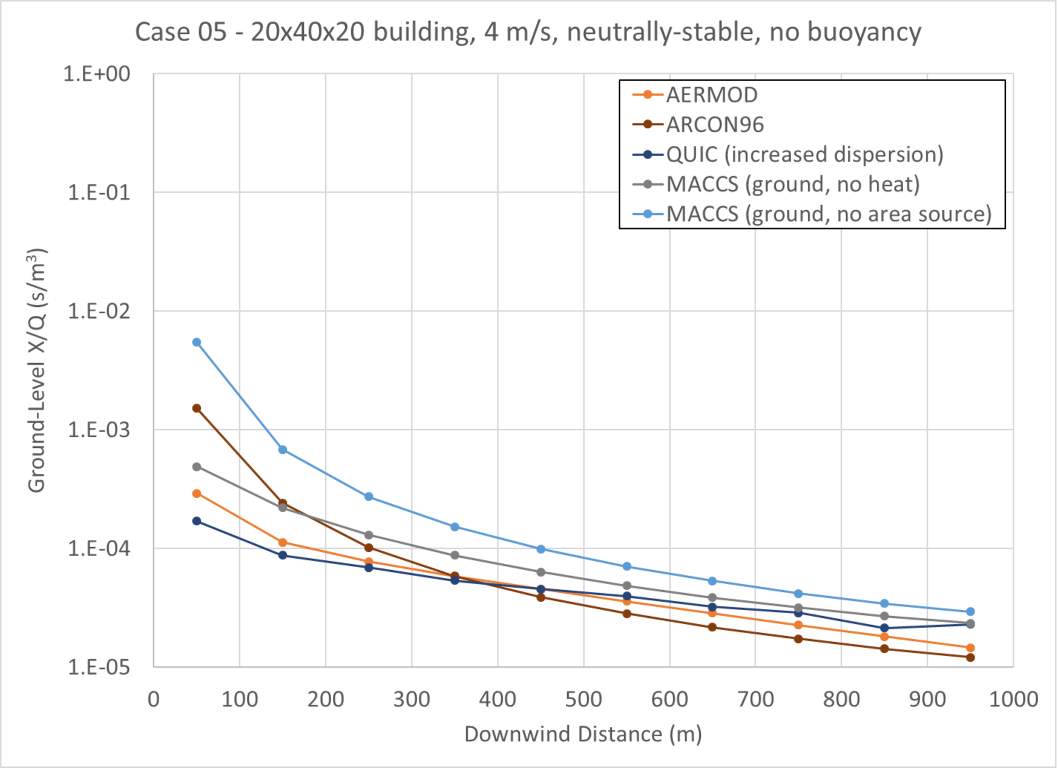
Need to **modify MACCS input** to **bound results** of other codes



MACCS 4.0 Nearfield Comparison Results with Updated Inputs

MACCS input **modified** to reflect a ground-level (1), non-buoyant (2) release (grey) **bounds AERMOD and QUIC** up to 1 km and **ARCON96** from 200 m up to 1 km

MACCS input **modified** to reflect a ground-level (1), non-buoyant (2), point-source (3) release (light blue) **bounds all three** up to 1 km



MACCS 4.1 Enhancements

Add two **new capabilities** in **MACCS 4.1** to facilitate **simulating** or **bounding** nearfield calculations performed with **other codes:**

- Implemented **Ramsdell and Fosmire** wake and meander algorithms used in ARCON96
- Updated existing meander model to fully implement wake and meander model equations from **US NRC Regulatory Guide 1.145** as implemented in PAVAN

Maintain existing MACCS capabilities to bound results with AERMOD and QUIC

New MACCS 4.1 Algorithms

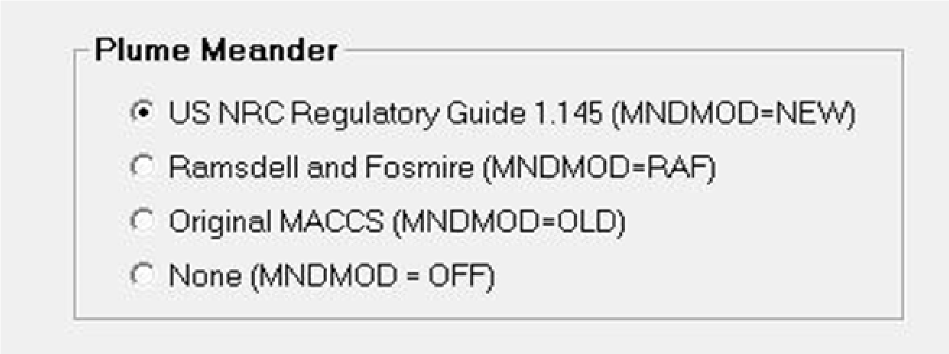
Ramsdell and Fosmire meander model used in ARCON96

US NRC Regulatory Guide 1.145 meander model as implemented in PAVAN

Ramsdell and Fosmire

$$\Sigma_y = (\sigma_y^2 + \Delta\sigma_{y1}^2 + \Delta\sigma_{y2}^2)^{1/2}$$

$$\Sigma_z = (\sigma_z^2 + \Delta\sigma_{z1}^2 + \Delta\sigma_{z2}^2)^{1/2}$$



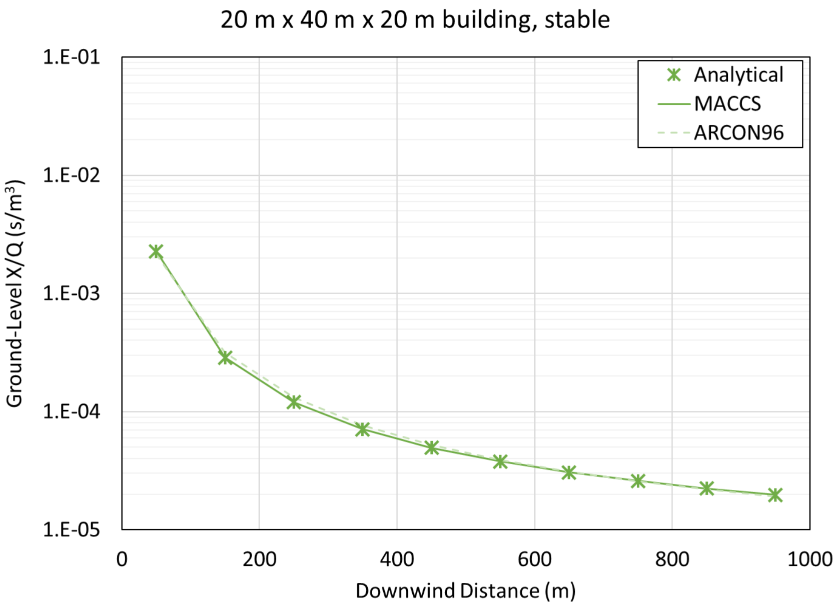
Reg. Guide 1.145

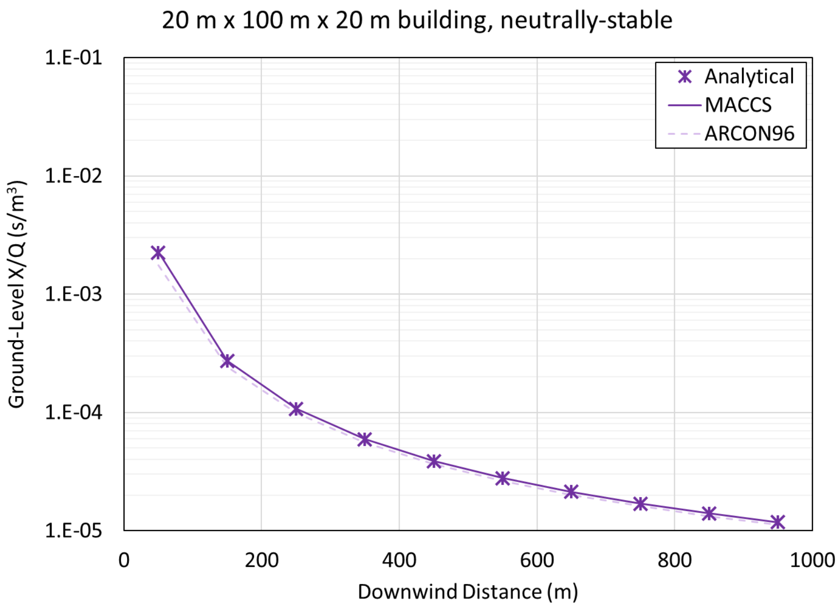
$$\chi/Q = \frac{1}{\bar{\bar{U}}_{10}(\pi\sigma_y\sigma_z + A/2)} \qquad (1)$$

$$\chi/Q = \frac{1}{\bar{\bar{U}}_{10}(3\pi\sigma_y\sigma_z)} \qquad (2)$$

$$\chi/Q = \frac{1}{\bar{\bar{U}}_{10}\pi\Sigma_y\sigma_z} \qquad (3)$$

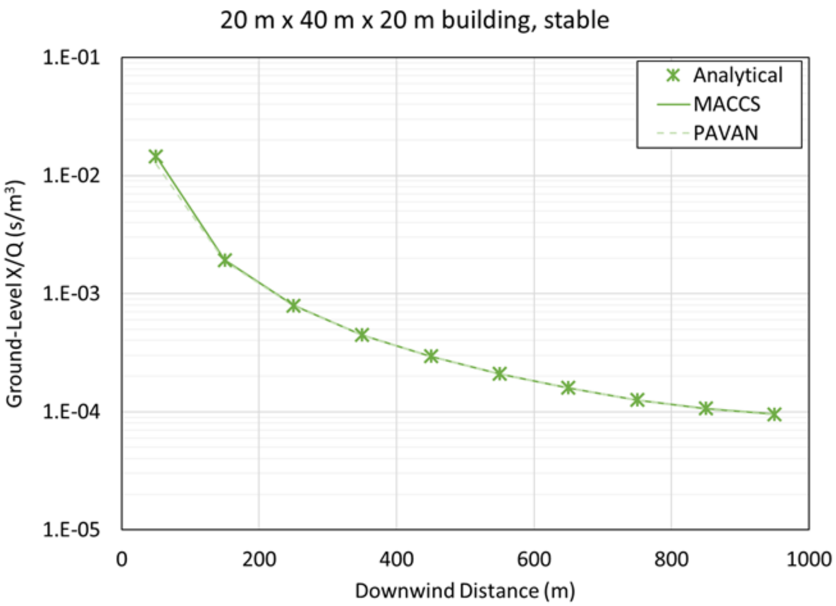
Verification-Ramsdell and Fosmire meander model

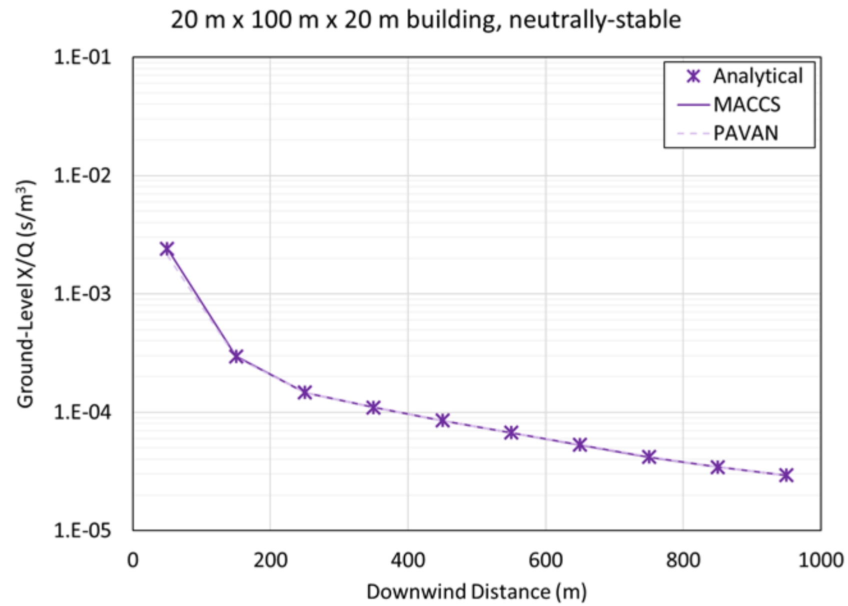




Generate results comparable to those from ARCON96 with MACCS when using the Ramsdell and Fosmire meander model

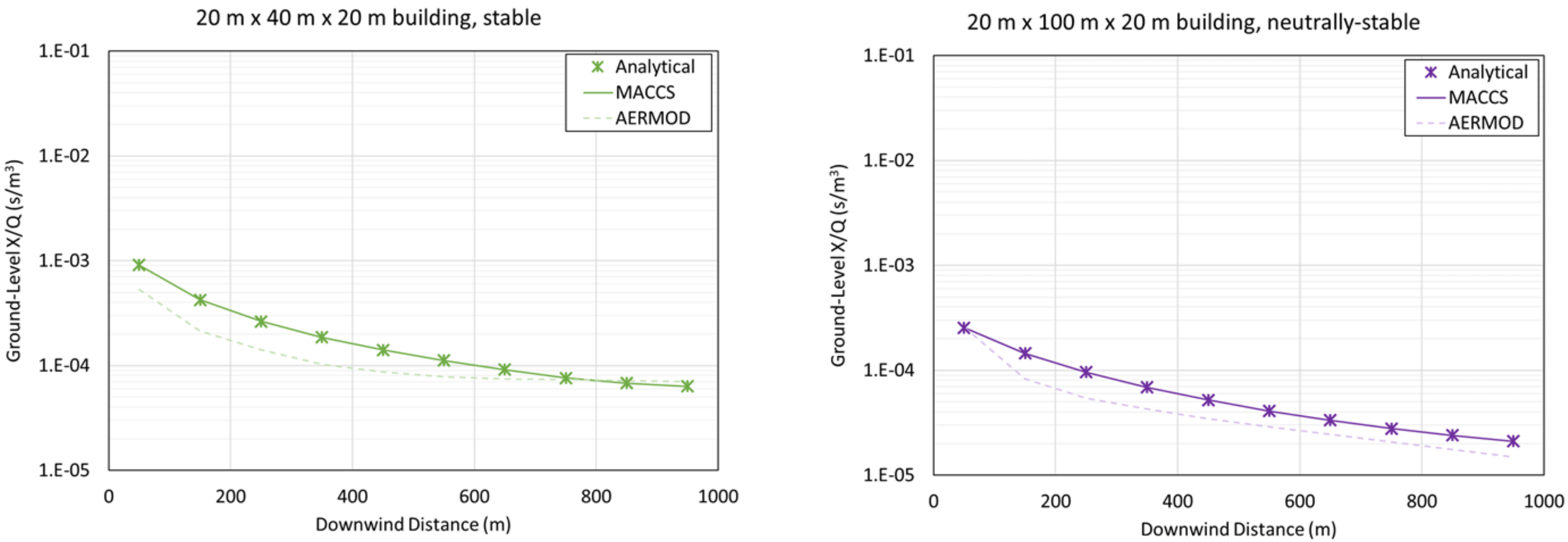
Verification-US NRC Reg Guide 1.145 meander model as implemented in PAVAN





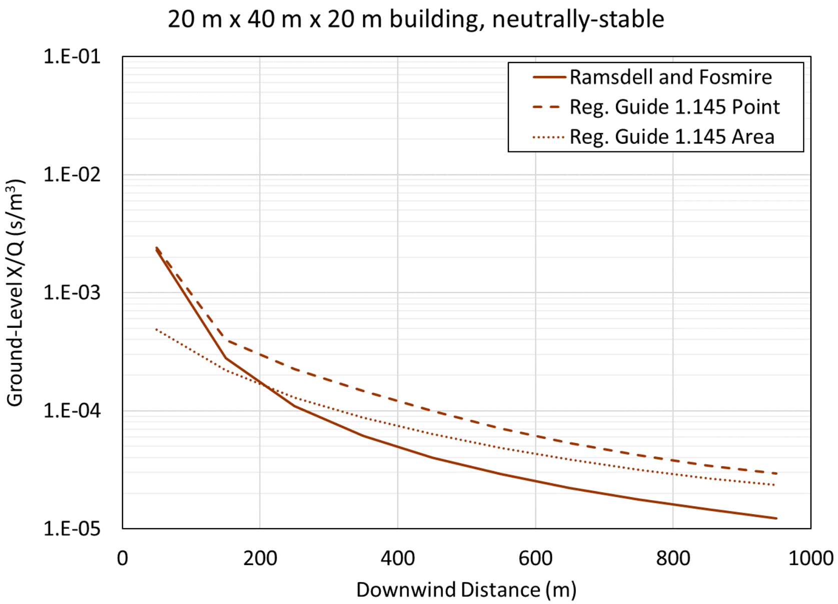
Generate results comparable to those from PAVAN with MACCS when using the full US NRC Regulatory Guide 1.145 meander model

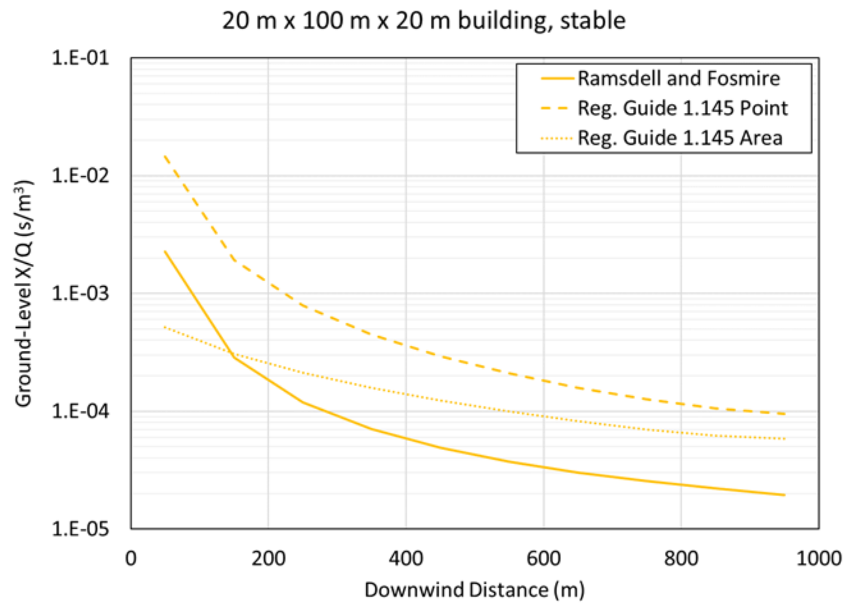
Verification-US NRC Reg Guide 1.145 meander model as implemented in MACCS 4.0



Maintain capability to bound AERMOD and QUIC results using recommended MACCS parameter choices

Model Comparisons (1/2)

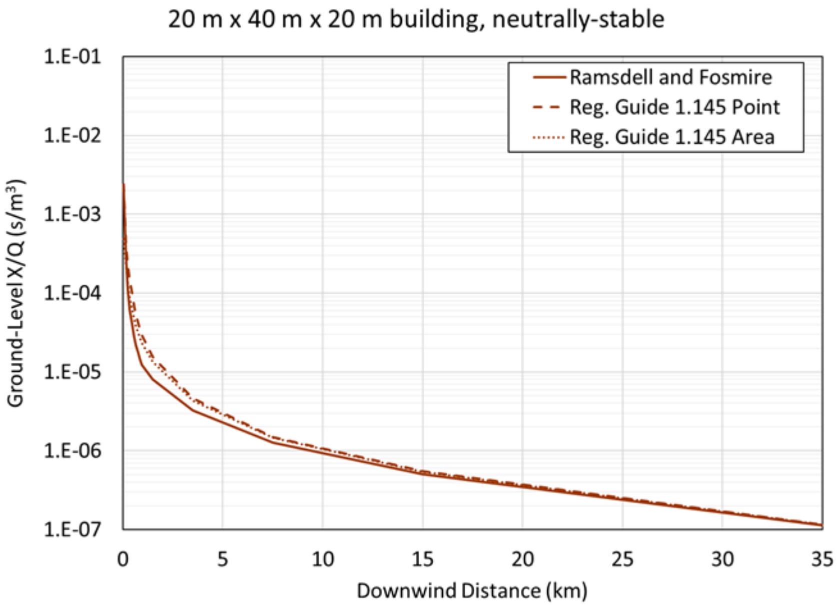


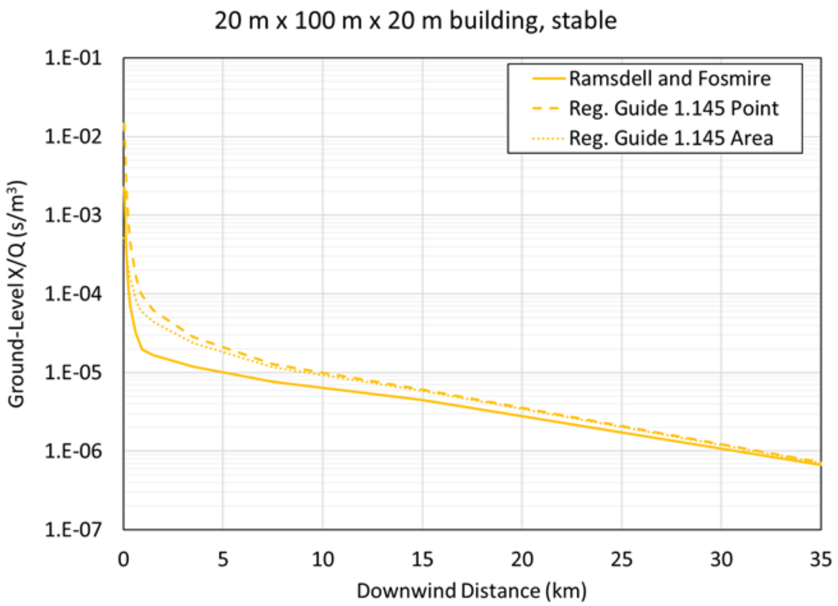


When using the full **US NRC Regulatory Guide 1.145 meander model**, the X/Q values for the test cases are **higher** than for the other two models

The X/Q values for the test cases with **MACCS Ramsdell and Fosmire plume meander model** are lower than the other two models except at distances of less than 200-300 m

Model Comparisons (2/2)





The three models converge with differences on the order of 5-10% at a distance of 35 km.

Summary Assessment of MACCS 4.0

ARCON96, AERMOD, and QUIC selected for **comparison** with **MACCS 4.0** based on initial evaluation

Based on the comparison, **MACCS 4.0 can be used in a conservative manner** at distances significantly shorter than 500 m downwind from a containment or reactor building

However, the MACCS user needs to **select** the MACCS input **parameters appropriately** to generate results that are adequately conservative for a specific application

Additional information available from final technical report (Clayton D.J and N.E. Bixler, "Assessment of the MACCS Code Applicability for Nearfield Consequence Analysis" SAND2020-2609, Sandia National Laboratories, Albuquerque, NM, February 2020, ADAMS Accession Number ML20059M032)

Summary of New MACCS 4.1 Capabilities

Additional **nearfield meander models** are **included** with **MACCS 4.1**

- Generate results comparable to those from ARCON96 with MACCS when using the Ramsdell and Fosmire meander model
- Generate results comparable to those from PAVAN with MACCS when using the full US NRC Regulatory Guide 1.145 meander model
- Maintain capability to bound AERMOD and QUIC results using recommended MACCS parameter choices

Comparing the plume meander model **results** shows

- When using the full **US NRC Regulatory Guide 1.145 meander model**, the X/Q values for the test cases are **higher** than for the other two models
- The X/Q values for the test cases with **MACCS Ramsdell and Fosmire plume meander model** are lower than the other two models except at distances of less than 200-300 m
- Beyond 1 km, **the three models converge** with differences on the order of 5-10% at a distance of 35 km.

MACCS 4.1 also available as **Linux** version
(see https://maccs.sandia.gov for more information)

Additional information available from final technical report (Clayton D.J, "Implementation of Additional Models into the MACCS Code for Nearfield Consequence Analysis" SAND2021-6924, Sandia National Laboratories, Albuquerque, NM, June 2021)

For questions or comments, please contact:

Daniel Clayton
MACCS Principal Investigator
Sandia National Laboratories
djclayt@sandia.gov

Keith Compton
Technical Monitor
U.S. Nuclear Regulatory Commission
Keith.Compton@nrc.gov

Backup slides

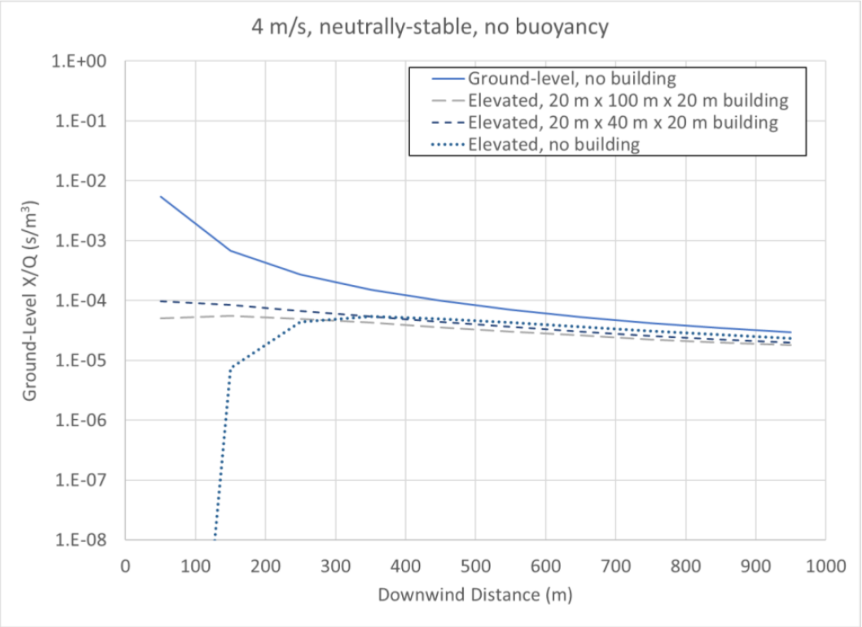
MACCS 4.0 Results

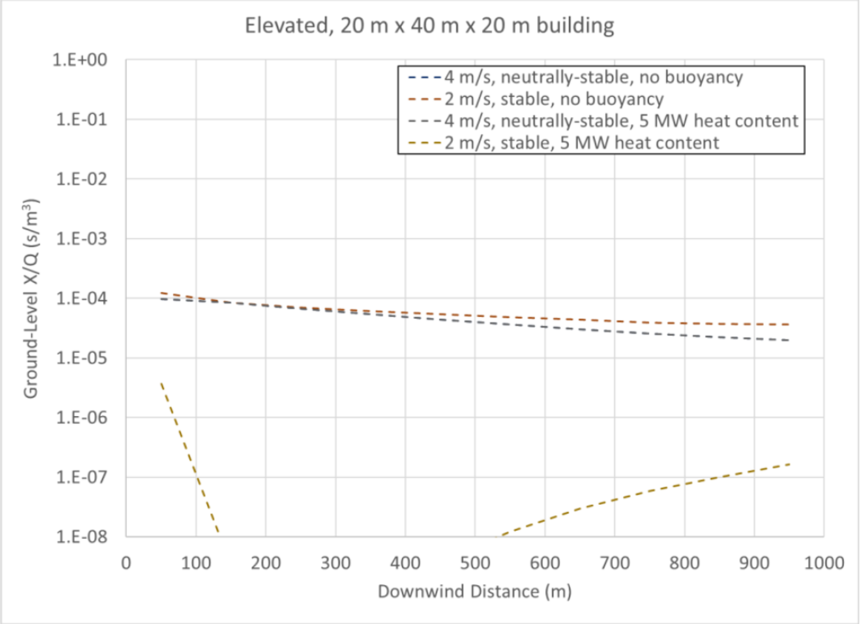
Building and **elevation** effects greatly **diminished** at 800 m **downwind**

Building significantly **increases dispersion** at short distances

Dilution for **stable** conditions generally **higher** than the corresponding dilution for **neutrally-stable** conditions

Buoyant plumes that escape building wake produce significantly **lower dilution values** due to fast plume rise compared with dispersion





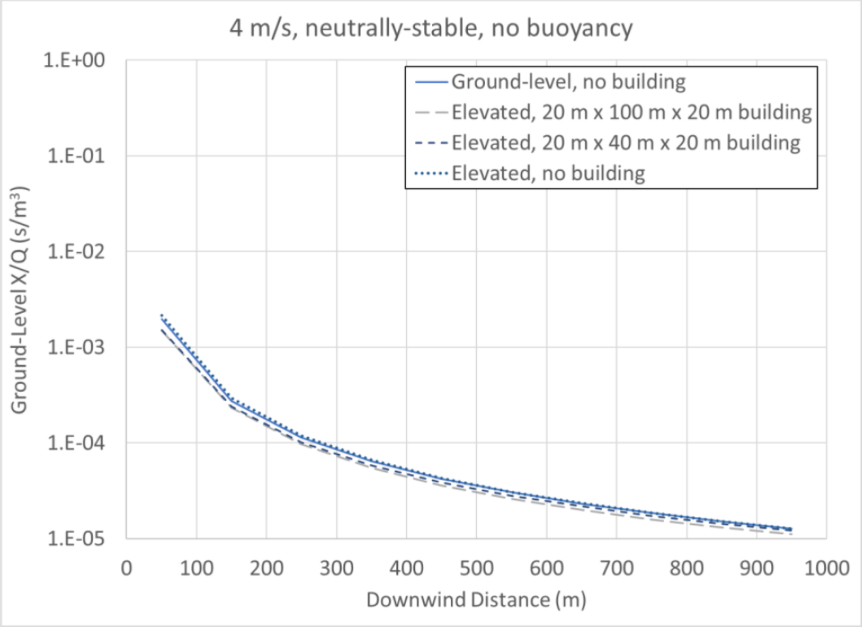
21

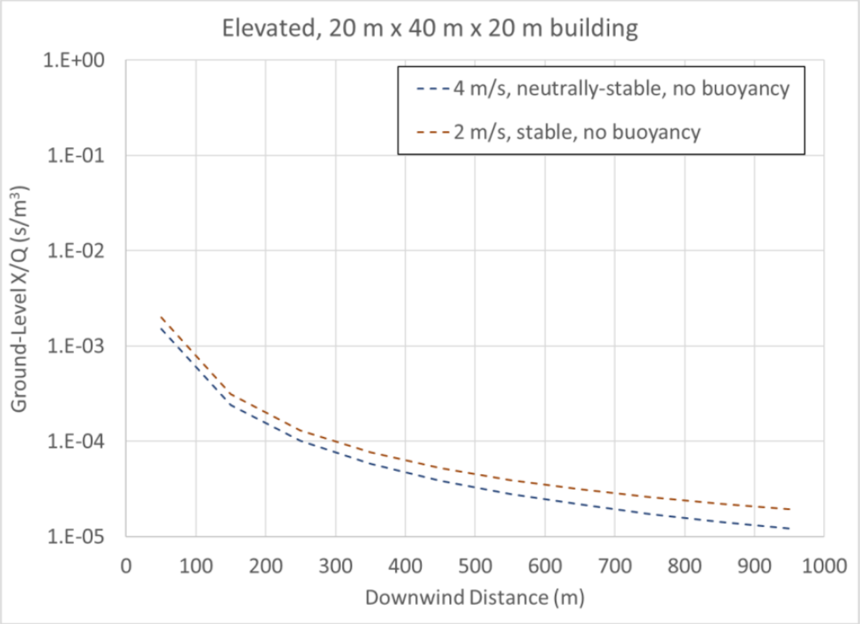
ARCON96 Results

Minimal change due to inclusion of **building** or **elevated** release within 1 km

Dilution for **stable** conditions generally **higher** than the corresponding dilution for **neutrally-stable** conditions

No plume rise model implemented; buoyant cases were not modeled





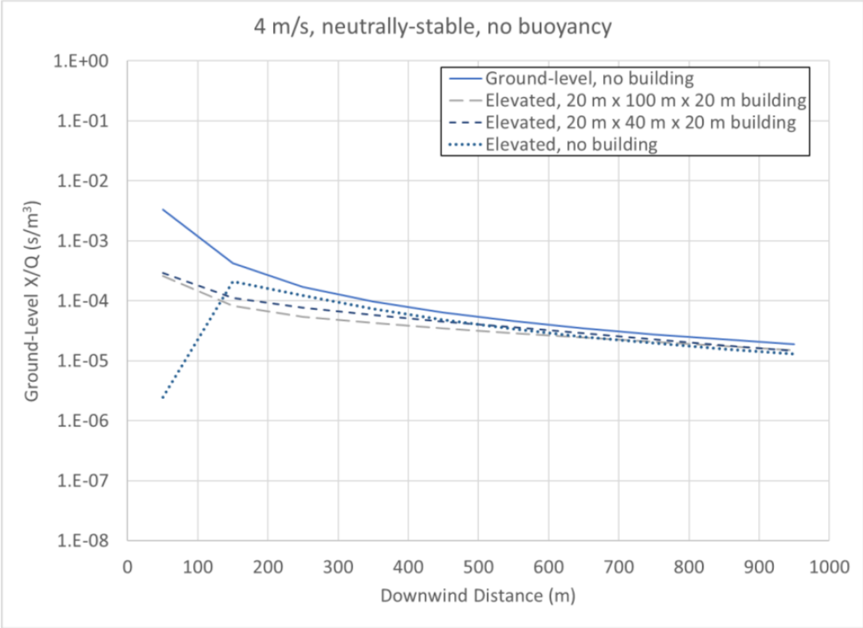
AERMOD Results

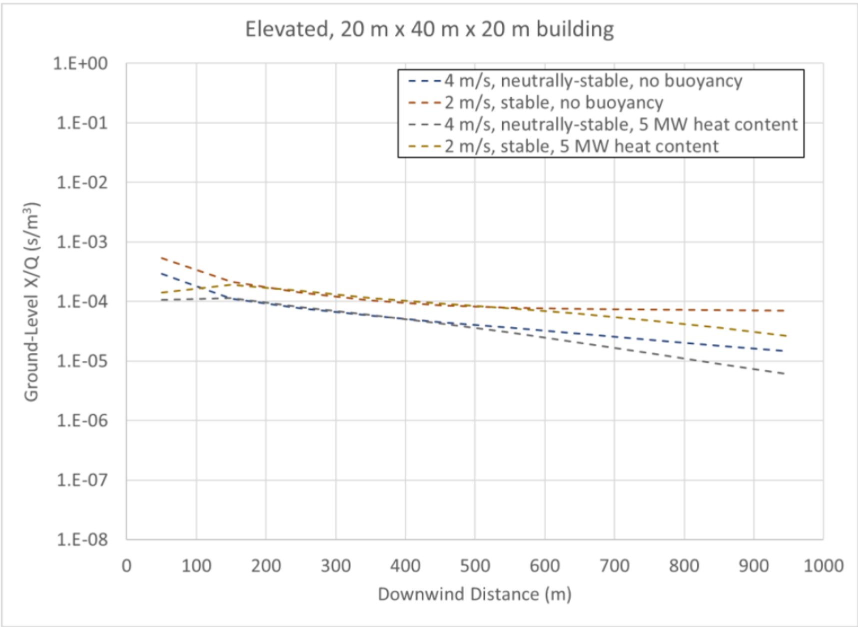
Building and **elevation** effects greatly **diminished** at 500 m **downwind**

Building significantly **increases dispersion** at short distances

Dilution for **stable** conditions generally **higher** than the corresponding dilution for **neutrally-stable** conditions

Minor differences due to **buoyancy**





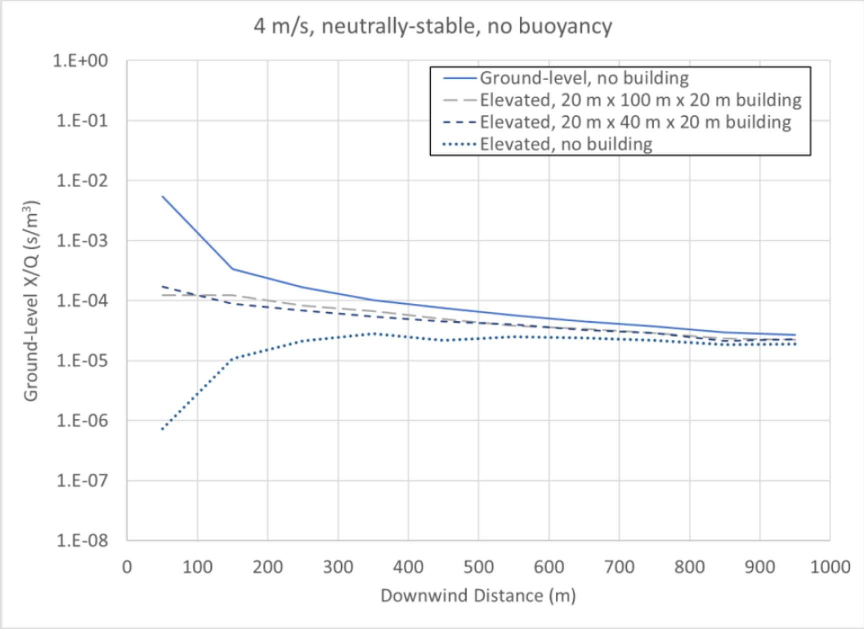
QUIC Results (1/2)

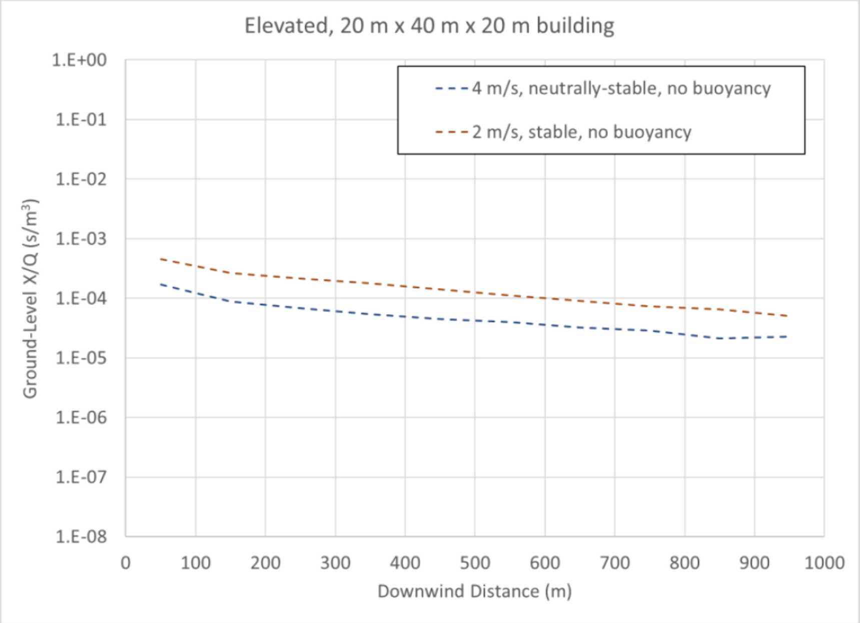
Building and **elevation** effects greatly **diminished** at 1 km **downwind**

Building significantly **increases dispersion** at short distances

Dilution for **stable** conditions generally **higher** than the corresponding dilution for **neutrally-stable** conditions

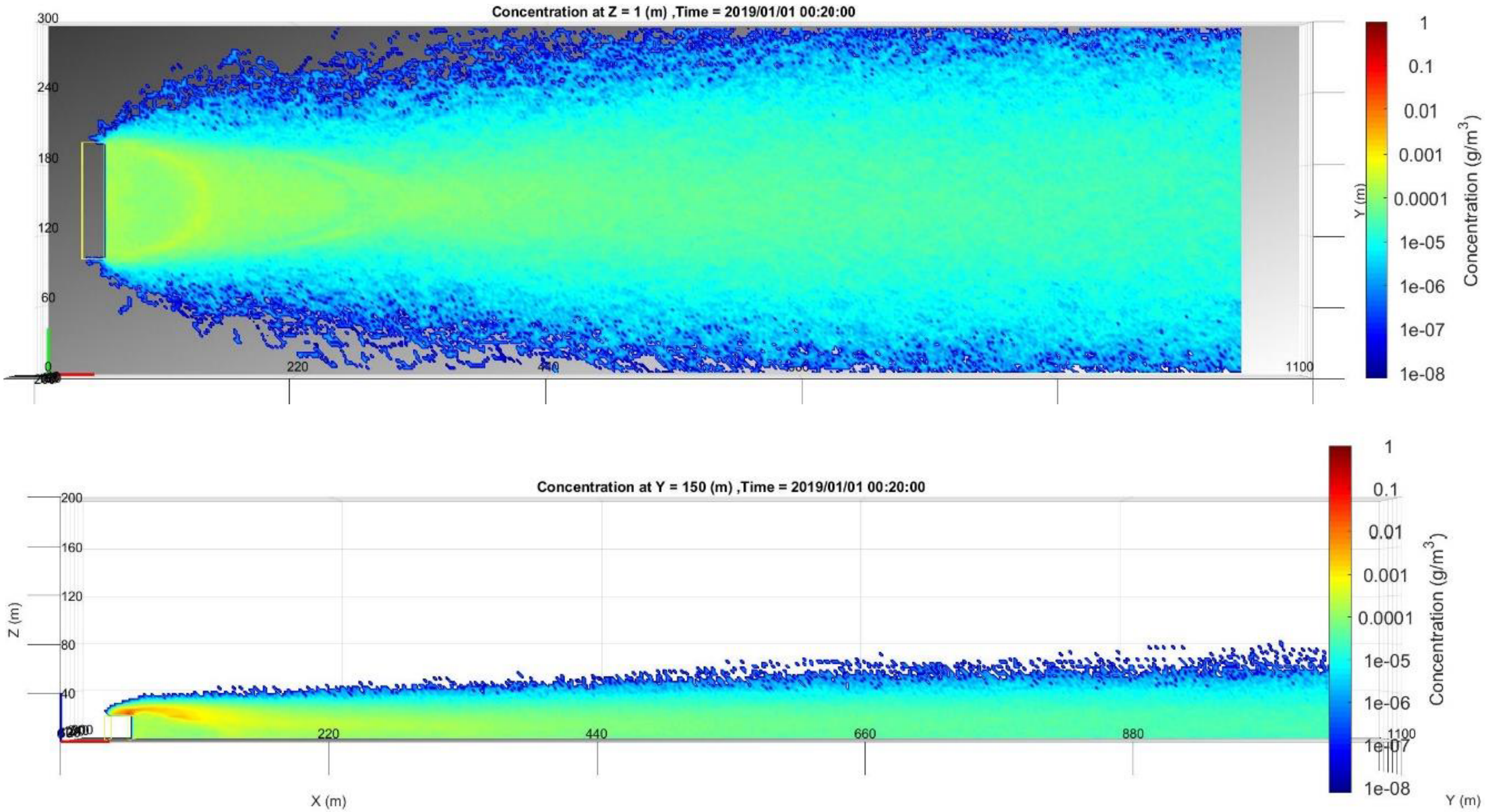
No straightforward way to implement buoyancy; buoyant cases were not modeled





QUIC Results (2/2)

Horizontal and vertical slices for a 4 m/s, neutrally-stable weather condition with a non-buoyant, elevated release from a 20 m x 100 m x 20 m building (Case 01)



Potential Modifications to MACCS Input

1. Specify a **ground-level release**, instead of a release at the height of the building
   - **ARCON96** model showed **little dependence on elevation** of release
   - **Wake-induced building downwash** observed in QUIC output
   - **Regulatory Guide 1.145** discusses releases less than 2.5 times building height should be modeled as **ground-level releases**
2. Specify **no buoyancy** (plume trapped in building wake)
   - **AERMOD** model showed **little dependence on buoyancy**
3. If **additional conservatism** needed or desired, model as a **point source**
   - **ARCON96** model showed **little dependence on building** size
   - **DOE** approach used for **collocated workers**
   - If point source **too bounding**, use an **intermediate building** wake size

Draft Interim Staff Guidance for the Safety Review of Light-Water Power Reactor Construction Permit Applications

Carolyn Lauron
New Reactor Licensing Branch (NRLB)
Division of New and Renewed Licenses (DNRL)
Office of Nuclear Reactor Regulation (NRR)

What is the purpose of today's presentation?

To facilitate stakeholder understanding of the information contained in the construction permit interim staff guidance recently noticed in the *Federal Register* for comment. (86 FR 71101)

This presentation should aid in the development and submission of stakeholder written comments consistent with the instructions in the *Federal Register* notice.

Why was the interim staff guidance developed?

- NRC anticipates the submission of construction permit applications.
- NRC last reviewed and issued a light-water power-reactor construction permit in the 1970s.
- Recently, NRC reviewed and issued licenses using the one-step process in 10 CFR Part 52.
- There are ongoing NRC activities to realign the requirements in 10 CFR Parts 50 and 52, and to develop guidance for non-light-water reactor designs.

Availability of Draft ISG DNRL-ISG-2022-XX

On December 14, 2021, the NRC published a notice in the *Federal Register* requesting comments on the draft interim staff guidance by January 28, 2022. (86 FR 71101)

The draft interim staff guidance may be found in the NRC's Agencywide Documents Access and Management System at this link: ML21165A157

Scope of Draft ISG DNRL-ISG-2022-XX

The scope of the interim staff guidance is the safety review of light-water power-reactor construction permit applications.

The interim staff guidance supplements the existing review guidance for light-water power-reactor applications found in NUREG-0800.

Parts of Draft ISG DNRL-ISG-2022-XX

- Main Body of Document
  - Purpose, Background, Rationale, Applicability
  - Guidance
  - Implementation
  - Backfitting and Issue Finality Discussion, Congressional Review Act
  - Final Resolution
  - References
- Appendix

Guidance in Draft ISG DNRL-ISG-2022-XX

Guidance Subsections

- Requirements for a Power Reactor Construction Permit Application
- Light-Water-Reactor Safety Review Guidance
- Special Topics
  - Relationship between the Construction Permit and Operating License reviews
  - Purposes and benefits of preapplication activities
  - Lessons learned from recently issued construction permits
  - Approach for reviewing concurrent license applications and applications incorporating prior NRC approvals
  - Potential effect of ongoing regulatory activities on construction permit reviews and
  - Licensing requirements for byproduct, source, or special nuclear material.

Appendix to Draft ISG DNRL-ISG-2022-XX

- Supplements existing guidance in NUREG-0800
  - Reiterates the context, expected engagement, and review approach
  - Clarifies guidance for *selected safety-related topics*
- Not intended to include all topics expected and reviewed in a construction permit application.

Clarifications in Appendix to Draft ISG DNRL-ISG-2022-XX

Select topics discussed:

- Siting
- Radiological Consequence Analyses
- Transient and Accident Analyses
- Structures, Systems, and Components
- Protective Coatings Systems
- Instrumentation and Control
- Electrical System Design and
- Radioactive Waste Management

Submitting Comments on DNRL-ISG-2022-XX

Link to *Federal Register notice:* 86 FR 71101

Two ways to submit comments:

1. **Federal Rulemaking Website**: Go to https://www.regulations.gov/ and search for **Docket ID NRC-2021-0162**.
   - Address questions about Docket IDs in Regulations.gov to Stacy Schumann; telephone: 301-415-0624; email: Stacy.Schumann@nrc.gov
   - For technical questions, contact Carolyn Lauron, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone: 301-415-2736, email: Carolyn.Lauron@nrc.gov
2. **Mail comments** to: Office of Administration, Mail Stop: TWFN-7-A60M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Program Management, Announcements and Editing Staff.

Questions and Answers

Advanced Reactor Stakeholder Public Meeting

Break

Meeting will resume at 1pm EST

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 323 588 045#

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

NUREG/CR-7289, “Nuclear Data Assessment for Advanced Reactors”

Advanced Reactor Stakeholder Meeting

January 19, 2022

NUREG/CR-7289
ORNL/TM-2021/2002

- [ADAMS Accession No. ML21349A369]

- Oak Ridge National Laboratory (ORNL)
  - F. Bostelmann
  - G. Ilas
  - C. Celik
  - A.M. Holcomb
  - W.A. Wieselquist

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

NUREG/CR-7289
ORNL/TM-2021/2002

NUCLEAR DATA ASSESSMENT FOR ADVANCED REACTORS

Office of Nuclear Reactor Regulation

Motivation/Background

Commercial Light Water Reactor Approach to Reactor Physics/Nuclear Design

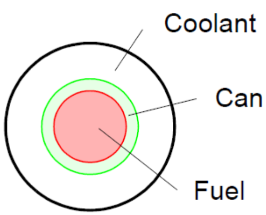
Data
(e.g., ENDF/B-VII.1)

↓

Cross Section Processing
(e.g., AMPX, NJOY)
Output: 100s of energy groups

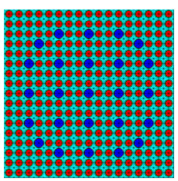
↓

1-D Pin Cell
(e.g., SCALE, CASMO)
Output: 20-100 energy groups



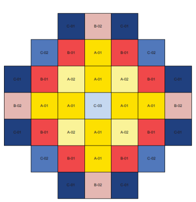
↓

2-D Assembly
(e.g., SCALE, CASMO)
Output: 2-4 Energy Groups,
Cross Section and Discontinuity Factors



↓

3-D Whole Core Simulator
(e.g., PARCS, SIMULATE)

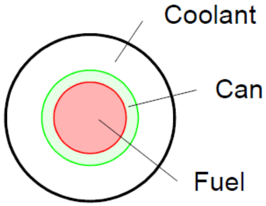


- *Start* with simplified geometry and detailed energy group structure, *End* with simplified group structure and 3D geometry
- Apply biases and uncertainties to calculated quantities of interest (QOIs):
  - Reactivity balance
  - Shutdown margin
  - Feedback coefficients
  - Power distribution

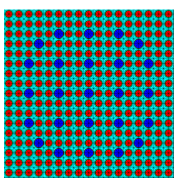
Commercial Light Water Reactor Approach to Reactor Physics/Nuclear Design

Data
(e.g., ENDF/B-VII.1)

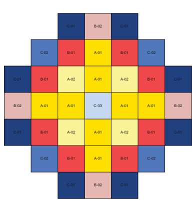
Cross Section Processing
(e.g., AMPX, NJOY)
Output: 100s of energy groups



1-D Pin Cell
(e.g., SCALE, CASMO)
Output: 20-100 energy groups



2-D Assembly
(e.g., SCALE, CASMO)
Output: 2-4 Energy Groups,
Cross Section and Discontinuity Factors



3-D Whole Core Simulator
(e.g., PARCS, SIMULATE)

- *Start* with simplified geometry and detailed energy group structure, *End* with simplified group structure and 3D geometry
- Apply biases and uncertainties to calculated quantities of interest (QOIs):
  - Reactivity balance
  - Shutdown margin
  - Feedback coefficients
  - Power distribution

Emphasized during safety review

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

Impact of Data Uncertainty

- QOIs verified via (1) startup physics testing, and (2) surveillance requirements

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ± 1% Δk/k of predicted values.

- Advanced Reactor examples*:
  - Changes in graphite data from ENDF/B-VII.0 to B-VII.1 (capture cross section) had a 1% Δk/k impact
  - No data for FLiBe/FLiNak thermal scattering, possible 2% Δk/k impact for thermal spectrum
- Uncertainties in nuclear data/physics modeling has the potential to adversely impact reactor operation

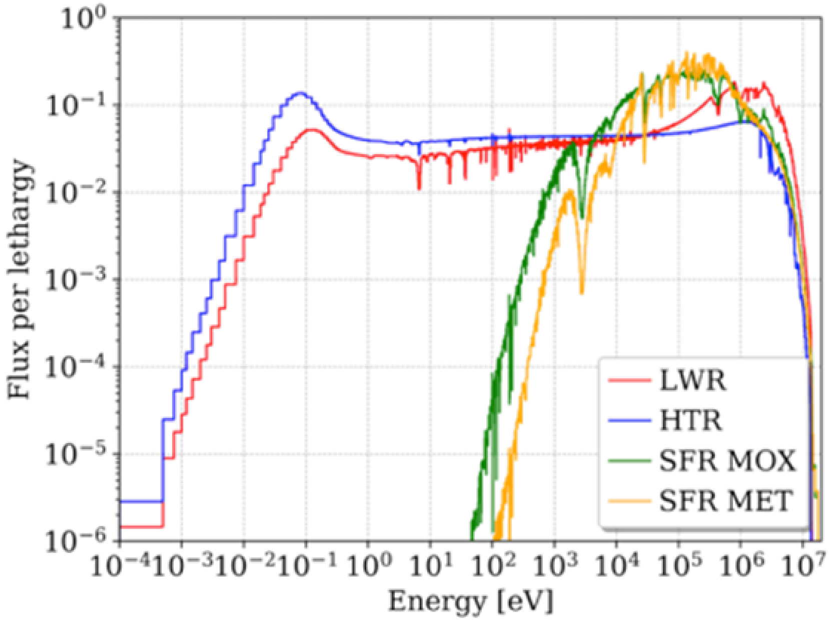
* Based on 2018 work performed at ORNL and available literature in 2019

Data Uncertainty and Licensing

- NRC review of nuclear design expected to emphasize uncertainty management
  - Appropriate application/justification of design margin into QOIs
  - Uncertainty update methodologies
  - Commitment to measurements/surveillances to verify design margin
  - Commitment to required actions in the event that measurements/surveillances fail to meet acceptance criteria

Data Challenges for Advanced Reactor Licensing

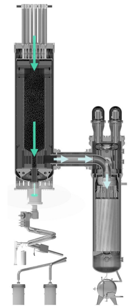
- Confidence in current nuclear data needs to be confirmed for non-LWRs:
  - Unique materials and neutron energy spectra
  - Nontraditional fuel forms
  - Limited integral validation data
- Nuclear data expertise:
  - Gaps in current nuclear data libraries?
  - Impact of gaps/uncertainties on QOIs?



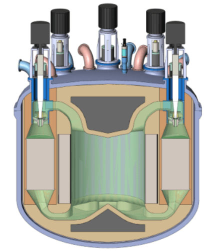
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Overview of NUREG/CR-7289, “Nuclear Data Assessment for Advanced Reactors”

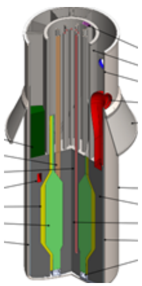
Technologies Considered



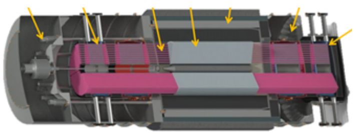
High Temperature Gas Reactor



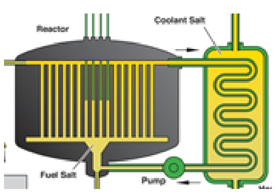
Molten Chloride Fast Spectrum Reactor



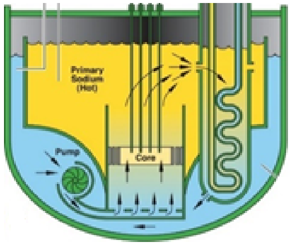
Fluoride Salt-Cooled High Temperature Reactor



Heat Pipe Microreactor



Graphite Moderated Molten Salt Reactor

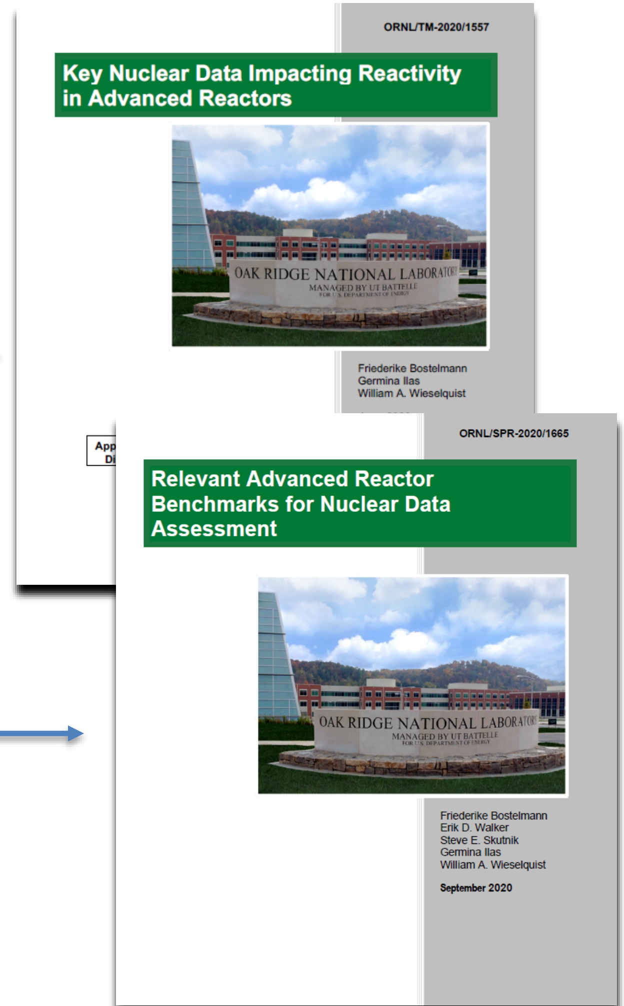


Sodium-Cooled Fast Reactor

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

Approach

- **4 Phases:**
  - **Phase 1 and 2**: Identify and assess key data impacting reactivity in non-LWRs based on literature review
  - **Phase 3**: Identify relevant benchmarks
  - **Phase 4:** Assess the impact of nuclear data uncertainty through propagation to key QOIs
    - Sensitivity and uncertainty analysis (performed using SCALE 6.3)



ADAMS Accession Nos. ML20274A052 and ML21125A256

Sensitivity and Uncertainty Analysis

Reactor technology	Selected benchmark[a]	Type
High Temperature Gas Reactor	HTR-10	Experiment
Fluoride Salt Cooled High Temperature Reactor	UC Berkeley Mark1 PB-FHR	Computational benchmark
Graphite-moderated Molten Salt Reactor	MSRE	Experiment
Heat Pipe Microreactor (metal-fueled)	INL Megapower Design A[b]	Computational benchmark
Sodium Cooled Fast Reactor (metal and oxide fueled)	EBR-II ABR-1000	Experiment Computational benchmark

[a] Although Fast Spectrum Molten Salt Reactors were identified as a relevant reactor concept, a concept with details sufficient for modeling could not be found in the open literature.

[b] The original design contains oxide fuel. However, for this project, metal fuel was assumed.

Sensitivity and Uncertainty Analysis

- Analyses were performed using ENDF/B-VII.0, ENDF/B-VII.1, and ENDF/B-VIII.0
- Sensitivity coefficients:
  - $S_{Y,\Sigma_{x,g}^{i}} = \frac{\Sigma_{x,g}^{i}}{Y}\frac{dY}{d\Sigma_{x,g}^{i}}$; (Y is the QOI, and $\Sigma_{x,g}^{i}$ is the data)
  - NUREG/CR-7289 reports sensitivity coefficients using ENDF/B-VII.1 (results using ENDF/B-VII.0 and ENDF/B-VIII.0 obtained values that are very close to ENDF/B-VII.1)

Results and Key Nuclear Data
(Subset of results from HTR-10 benchmark)
Nominal Results

Nominal Reactivity Impacts for QOIs

QOIs	ENDF/B-VII.0	ENDF/B-VII.1	ENDF/B-VIII.0	$\frac{\mathrm{VII.1}}{\mathrm{VII.0}} - 1$	$\frac{\mathrm{VIII.0}}{\mathrm{VII.1}} - 1$
Fuel temperature	-243 ± 22	-241 ± 25	-222 ± 25	3 ± 33	19 ± 36
Pebble gr. density	1182 ± 23	1175 ± 23	1201 ± 27	-8 ± 32	26 ± 35
Pebble gr. impurities	-602 ± 23	-623 ± 23	-588 ± 25	-21 ± 32	35 ± 34
Pebble gr. temperature	-1948 ± 23	-1960 ± 22	-1701 ± 25	-11 ± 32	259 ± 33
Structural gr. density	546 ± 25	504 ± 22	543 ± 24	-43 ± 33	40 ± 32
Structural gr. impurities	-3947 ± 26	-3877 ± 25	-3807 ± 25	70 ± 36	70 ± 35
Structural gr. temperature	780 ± 24	783 ± 22	798 ± 24	4 ± 33	14 ± 33

Results and Key Nuclear Data
(Subset of results from HTR-10 benchmark)
Sensitivity Analysis Results

Key Nuclear Data Impacting Pebble Graphite Temperature Feedback

Nuclide	Reaction	Sensitivity (reducing negative $\Delta\rho$)	Nuclide	Reaction	Sensitivity (increasing negative $\Delta\rho$)
u-235	fission	1.196e+00 ± 6.070e-03	b-10	n,α	-9.273e-02 ± 1.440e-03
u-235	$\bar{\nu}$	9.976e-01 ± 6.552e-04	u-238	n,γ	-3.655e-02 ± 1.764e-03
s-28	elastic	9.796e-03 ± 6.801e-03	n-14	n,p	-5.147e-03 ± 1.908e-04
c	elastic	9.083e-03 ± 9.656e-03	u-235	elastic	-3.560e-03 ± 3.272e-03
u-238	elastic	8.487e-03 ± 9.148e-03	si-28	n,γ	-4.577e-04 ± 2.769e-05
o-16	elastic	6.737e-03 ± 8.590e-03	graphite	n,α	-8.149e-04 ± 2.176e-04
u-235	n,γ	6.585e-03 ± 1.145e-03	si-28	n,n′	-3.930e-04 ± 4.912e-04
n-14	elastic	6.281e-03 ± 6.051e-03	n-14	n,γ	-2.084e-04 ± 7.821e-06
graphite	n,n′	4.702e-03 ± 2.311e-03	ar-40	elastic	-1.988e-04 ± 1.457e-04
u-238	nu-fission	2.402e-03 ± 6.552e-04	n-14	n,α	-4.236e-05 ± 1.867e-06

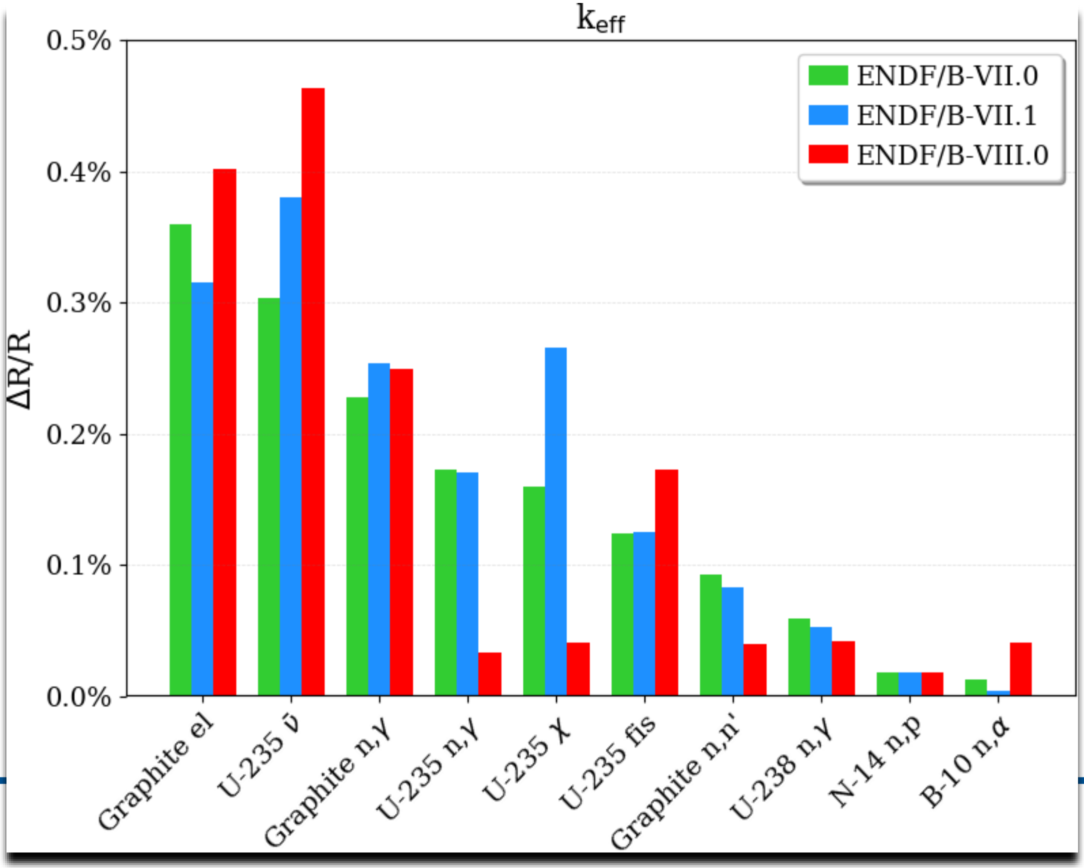
Results and Key Nuclear Data
(Subset of results from HTR-10 benchmark)
Uncertainty Analysis Results

Uncertainty in QOIs due to nuclear data

QOIs	ENDF/B-VII.0	ENDF/B-VII.1	ENDF/B-VIII.0	$\frac{\mathbf{VII.1}}{\mathbf{VII.0}} - 1$	$\frac{\mathbf{VIII.0}}{\mathbf{VII.1}} - 1$
k_{eff}	0.607%	0.668%	0.690%	10.1%	3.3%
Fuel temperature	1.124%	1.192%	1.030%	6.1%	-13.6%
Pebble gr. density	0.667%	0.848%	0.618%	27.1%	-27.1%
Pebble gr. impurities	0.639%	0.749%	1.126%	17.2%	50.3%
Pebble gr. temperature	0.694%	0.753%	0.972%	8.4%	29.1%
Structural gr. density	0.873%	0.952%	0.820%	9.1%	-13.9%
Structural gr. impurities	0.921%	1.109%	0.990%	20.3%	-10.7%
Structural gr. temperature	0.998%	1.135%	0.920%	13.7%	-18.9%

Results and Key Nuclear Data
(Subset of results from HTR-10 benchmark)
Uncertainty Analysis Results

Top Nuclear Data Contributors to Multiplication Factor Uncertainty



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

Conclusions

- Major data gaps from the libraries:
  - Thermal scattering kernel for molten salts
  - Uncertainty for thermal scattering (e.g., graphite)
  - Angular scattering uncertainty for fast spectrum reactors
- In general, the most important reactions were shown to be:
  - Neutron multiplicity, fission and radiative capture cross sections of fissile isotopes (e.g., U-235)
  - Radiative capture cross sections of fertile isotopes (e.g., U-238)
- Other significant contributors:
  - Capture cross sections of fission products*
  - Capture cross sections of neutron absorbing material (e.g., Gd or B)
  - Scattering reactions with the coolant and structural materials for fast spectrum systems
- For Molten Salt Reactors, in particular, additional neutron capture reactions such as (n,p) and (n,t) for salt components (e.g., Li and Cl) are significant contributors to the reactivity balance.

* Results of study with respect to depletion/burnup are limited due to (1) unavailability of benchmarks and relevant data, and (2) capability not currently available to fully propagate uncertainty in depletion analyses.

Conclusions

- Calculated uncertainty in reactivity balance due to nuclear data is generally greater than what is used in LWR nuclear design.
- Large uncertainties that are not considered relevant in LWRs studies were found to be significant for several advanced reactor systems:
  - All fast spectrum systems impacted by larger uncertainties in U-238 inelastic scattering and U-235 radiative capture at higher energies
  - A large uncertainty in the Li-7 capture cross section causes larger uncertainty in all QOIs for systems that use lithium as part of a salt coolant.
- No performance differences observed between the different libraries (i.e., ENDF/B-VII.0, ENDF/B-VII.1, and ENDF/B-VIII.0)
  - One exception being ENDF/B-VII.1 and ENDF/B-III.0 perform better for high temperature gas reactors because of the adjusted carbon capture cross section.
- NUREG/CR-7278 provides useful insight regarding nuclear design margins to accommodate gaps and uncertainty in the nuclear data.

SCALE and MELCOR development and application for non-LWRs

Advanced Reactor Stakeholder Meeting

January 19, 2022

NRC strategy for severe accident analysis


U.S.NRC
Revision 1
January 31, 2020
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

Volume 3
ML20030A178

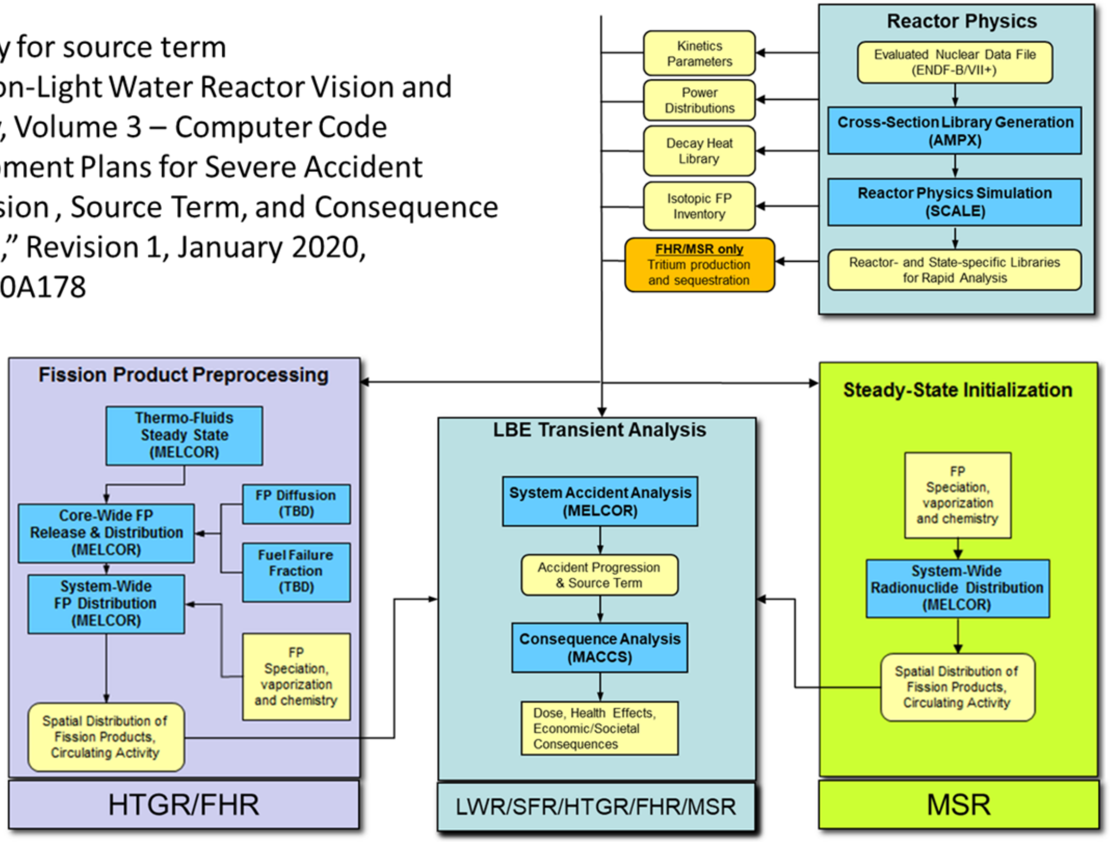
U.S.NRC
Revision 1
March 31, 2021
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

Volume 5
ML21088A047

Evaluation Model and Suite of Codes

Code strategy for source term

"NRC Non-Light Water Reactor Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis," Revision 1, January 2020, ML20030A178



scale

MELCOR

SCALE MELCOR Non-LWR Demonstration Project – objectives

Understand severe accident behavior and provide insights for regulatory guidance

Facilitate dialogue on staff's approach for source term

Demonstrate use of SCALE and MELCOR

- Identify accident characteristics and uncertainties affecting source term
- Develop publicly available input models for representative designs

SCALE MELCOR Non-LWR Demonstration Project – approach

1. Use SCALE to estimate core decay heat, radionuclide inventory, reactivity coefficients

2. Build MELCOR full-plant input model

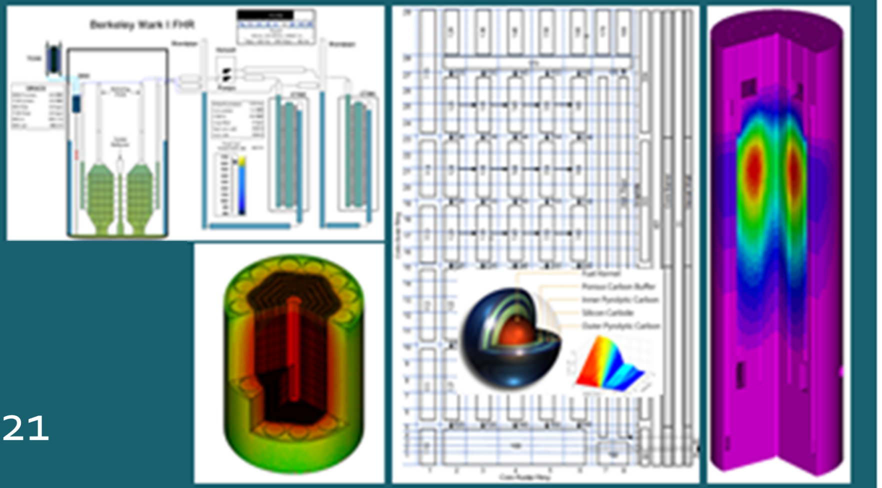
3. Select accident scenarios

4. Perform MELCOR simulations for the selected scenarios and debug
    - Base case
    - Sensitivity cases

5. Public workshops to discuss the modeling and sample results

Public Workshop: SCALE/ MELCOR Non LWR Source Term Demonstration Project

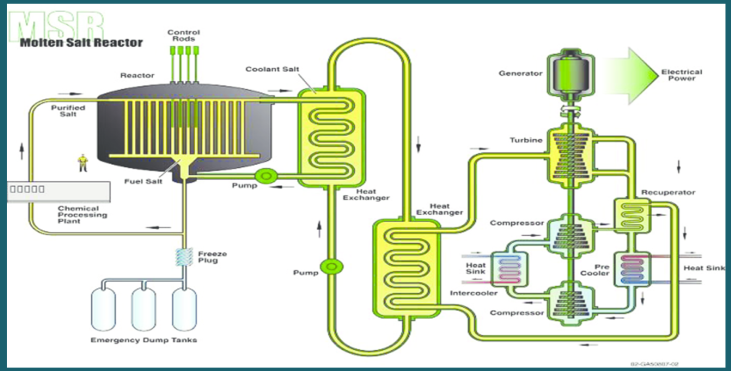
Heat pipe reactor – June 29, 2021
Gas cooled reactor – July 20, 2021
Pebble bed molten-salt-cooled reactor – Sept 14, 2021

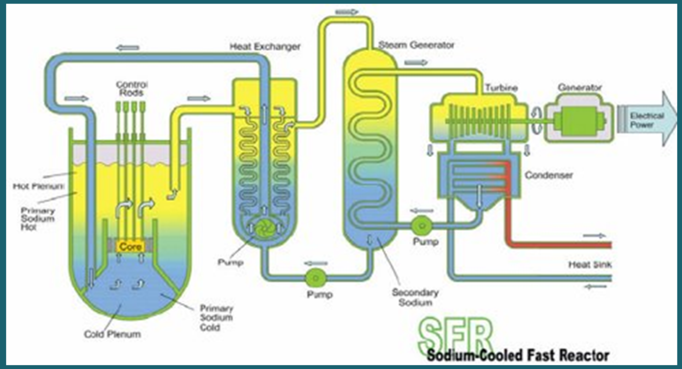


For More Information

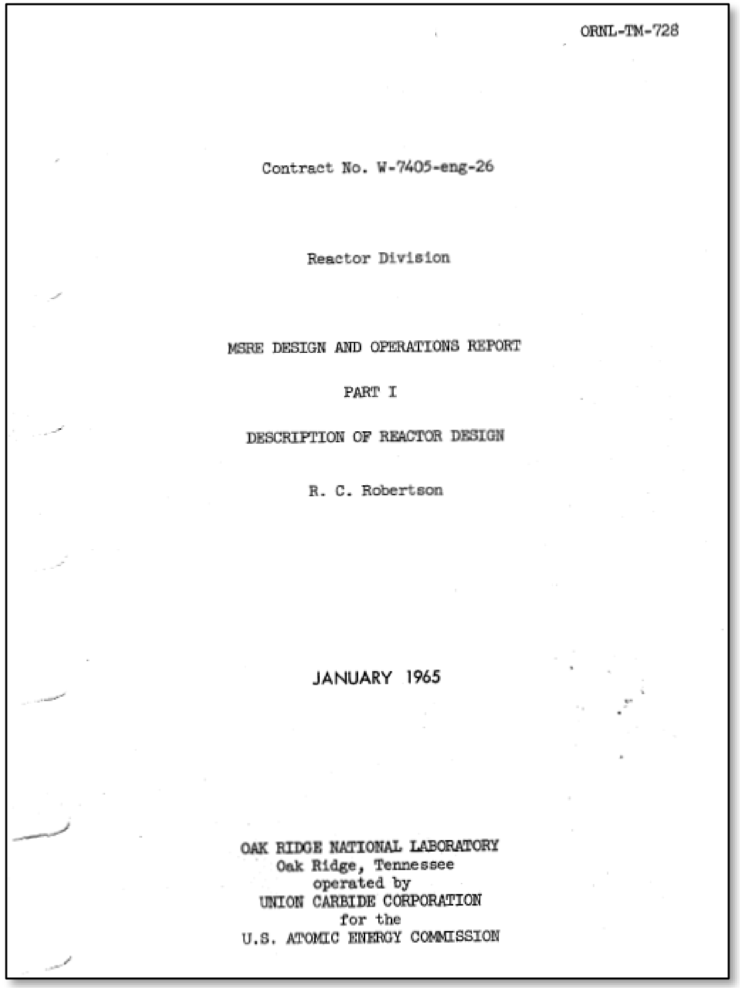
Coming in 2022

Molten-salt fueled reactor
Sodium-cooled fast reactor





Molten Salt Reactor Experiment (MSRE)



ORNL-TM-728

Contract No. W-7405-eng-26

Reactor Division

MSRE DESIGN AND OPERATIONS REPORT

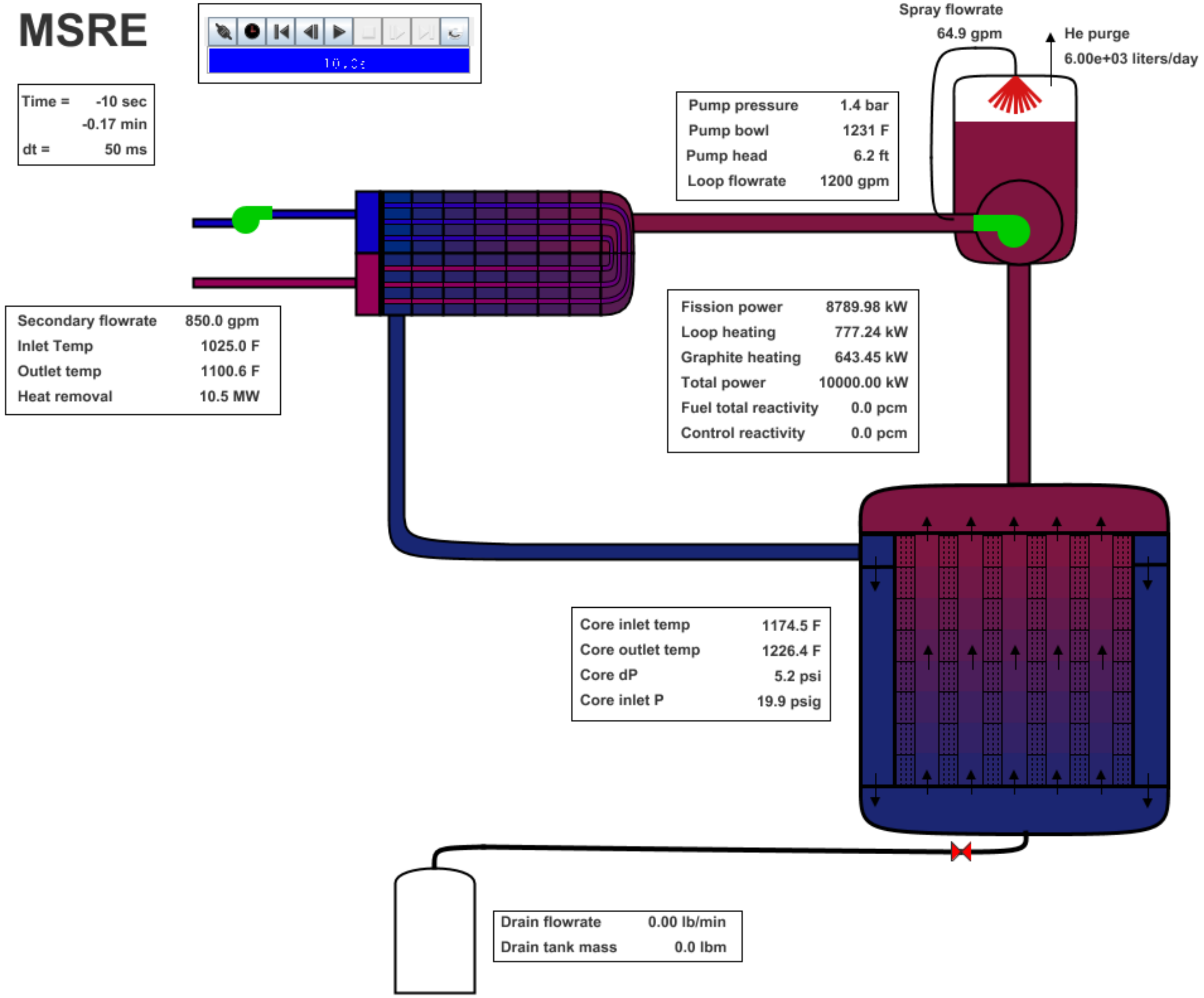
PART I

DESCRIPTION OF REACTOR DESIGN

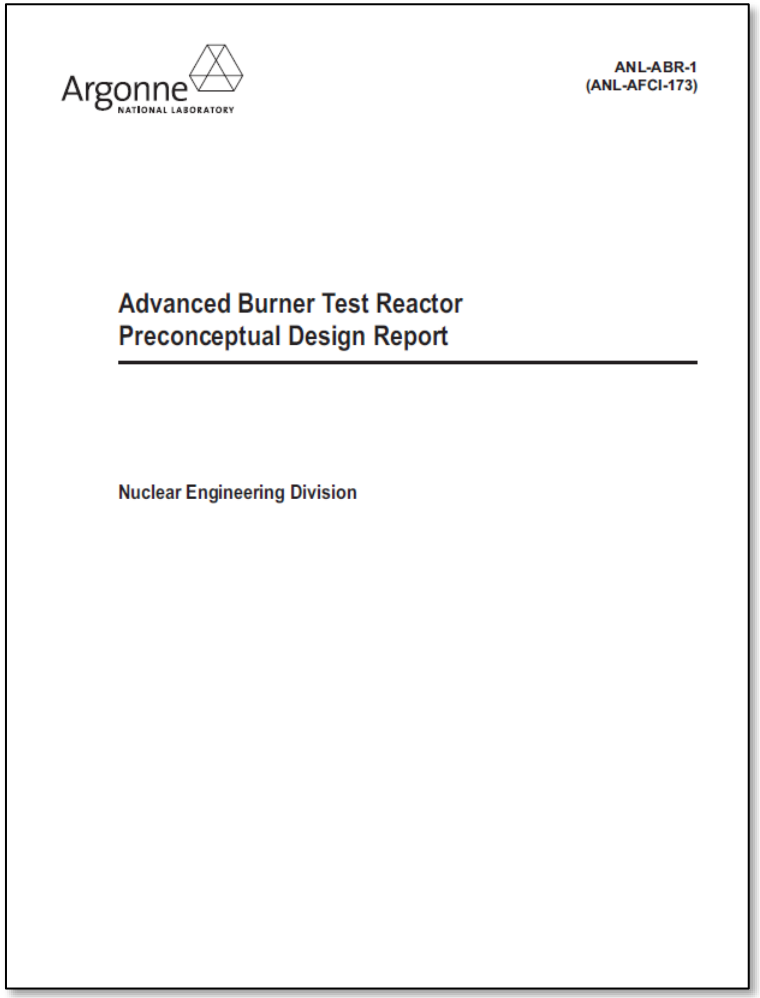
R. C. Robertson

JANUARY 1965

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION



Advanced Burner Test Reactor (ABTR)


Argonne NATIONAL LABORATORY

ANL-ABR-1
(ANL-AFCI-173)

Advanced Burner Test Reactor
Preconceptual Design Report

Nuclear Engineering Division

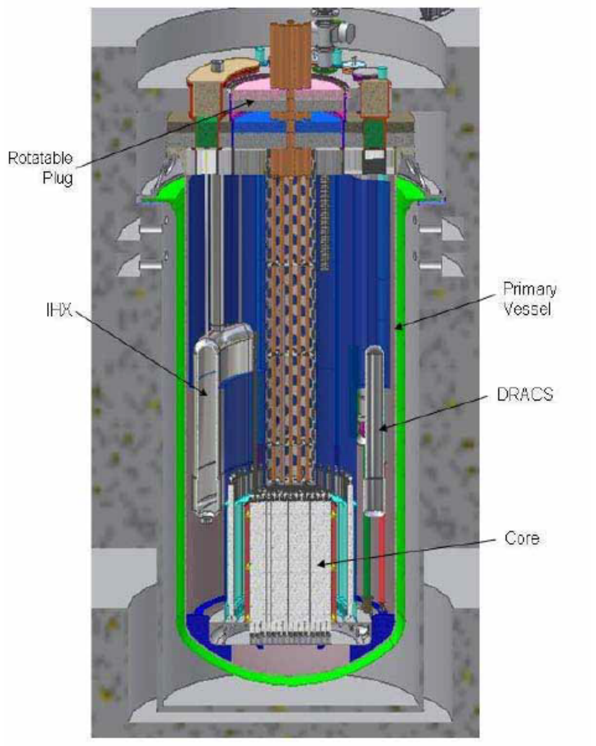


Figure 3 Schematic View of Primary System

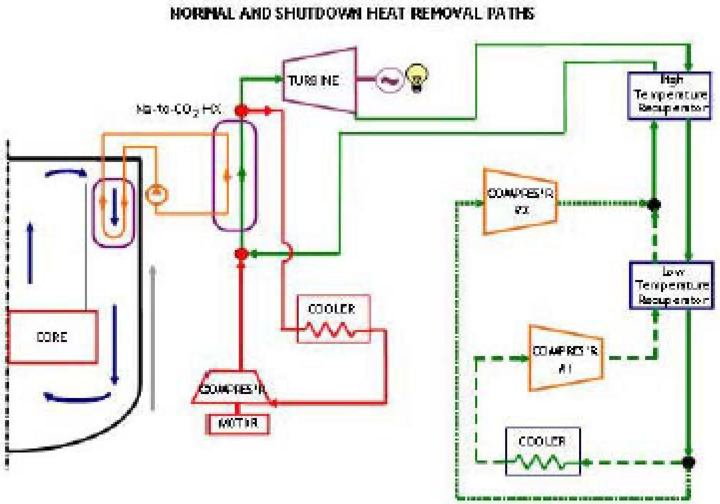


Figure 5 Overall Thermodynamic Cycle

SCALE analysis approach for MSR

3 models run in an iterative fashion to predict nuclide inventory, decay heat, and reactivity feedback coefficients at selected point in the operating cycle

Time snapshot

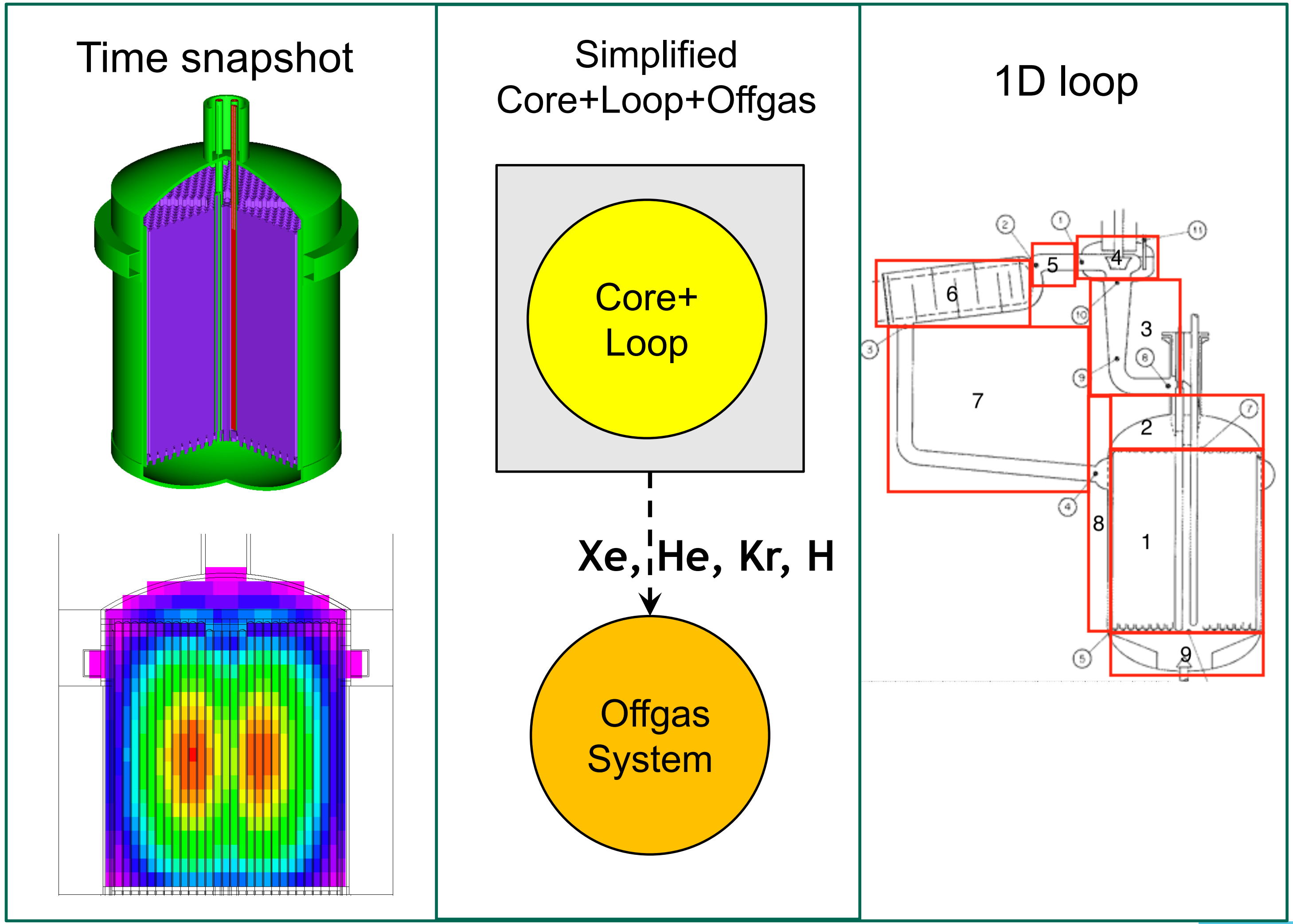
- predicts core neutron flux at a point in the operating cycle

Simplified core + loop + offgas

- predicts primary-system-average nuclide inventory over time

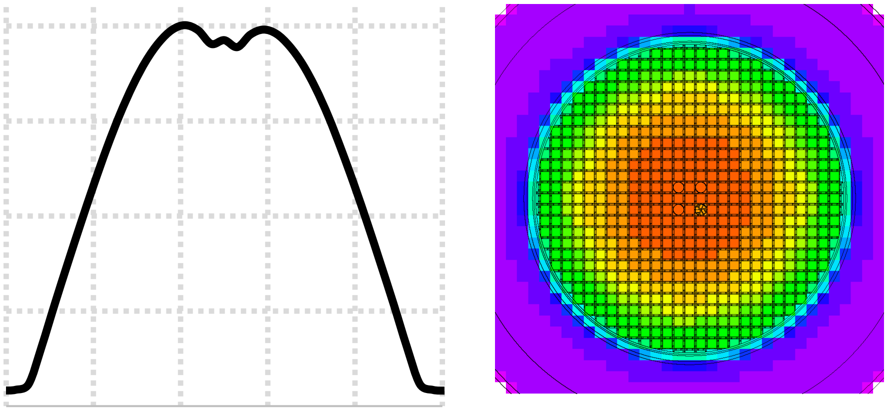
1D loop model

- predicts nuclide inventory in each section of the loop

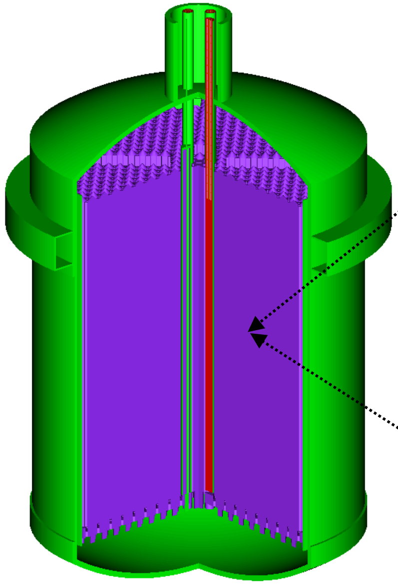


Time snapshot model

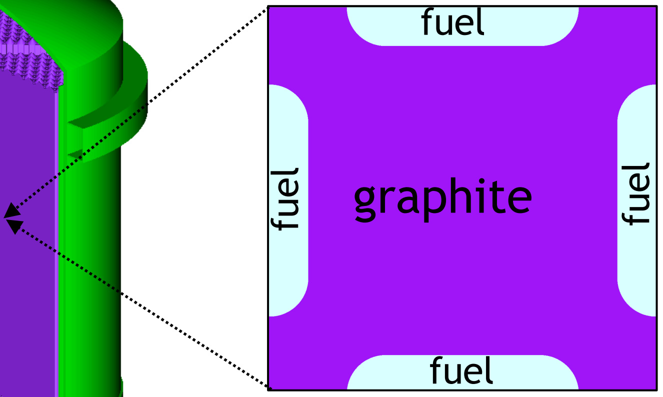
- Predicts 3D flux profiles via axial/radial discretization
  - Currently using 30 axial levels, 7 radial rings
- Investigating sensitivity of reactivity feedback to various modeling parameters



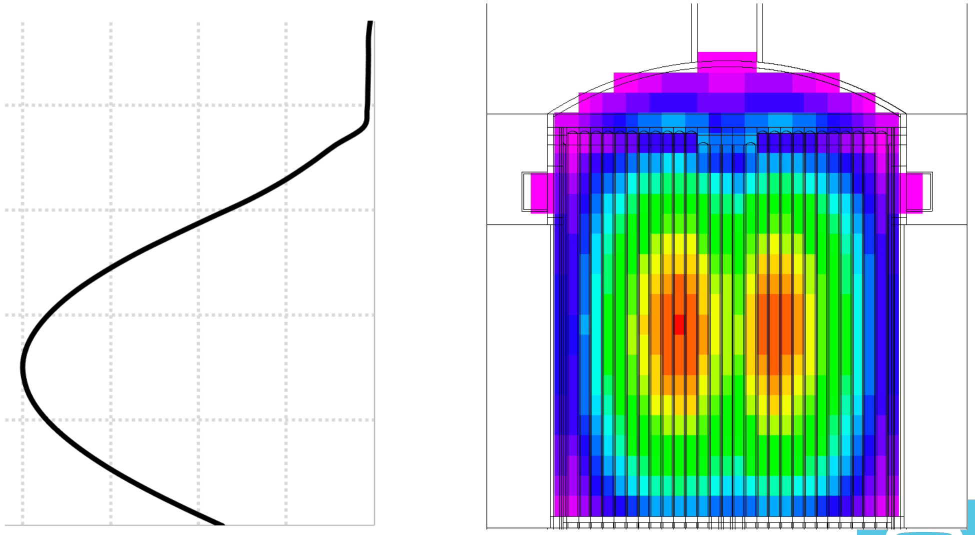
Radial flux distribution



SCALE 3D full core MSRE model



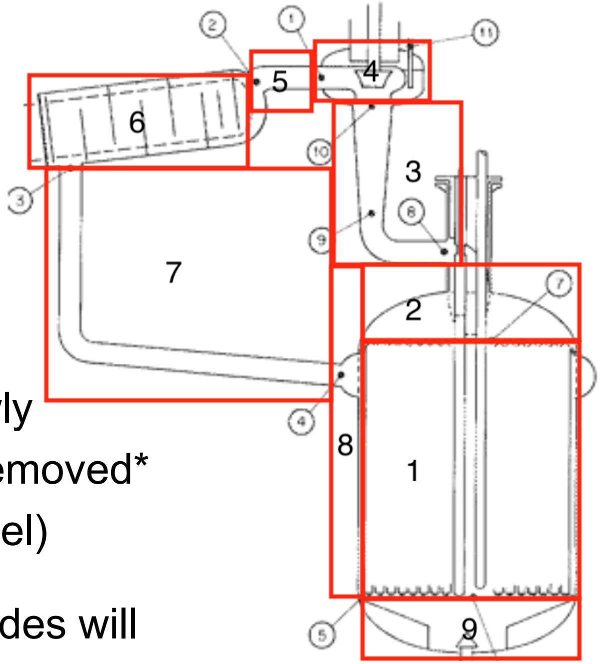
Cross section of unit cell



Axial flux distribution

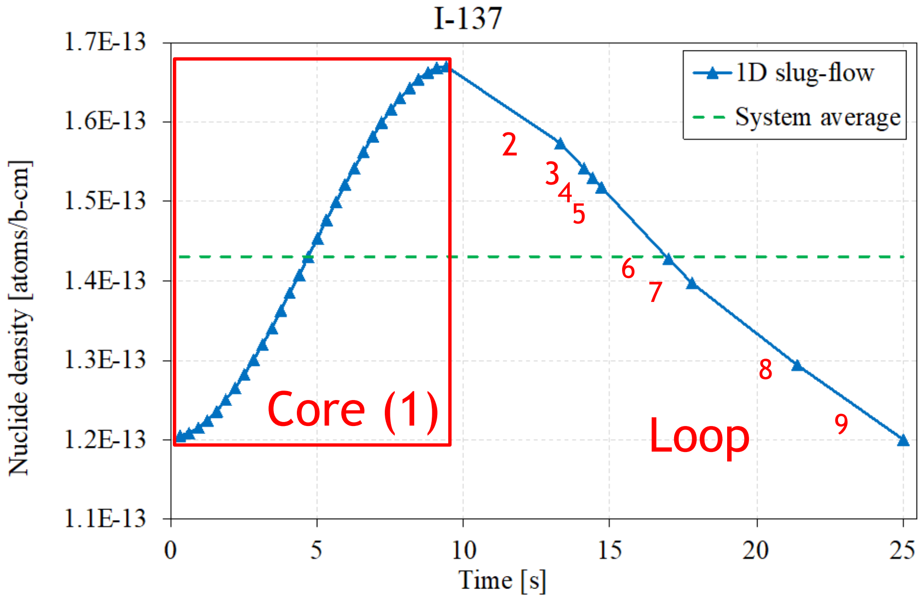
1D loop model

- Predicts nuclide inventory in each section of the loop
- As fuel salt travels the loop
  - Long-lived* nuclides will slowly accumulate/be removed* (same as solid fuel)
  - Short-lived* nuclides will oscillate about an equilibrium

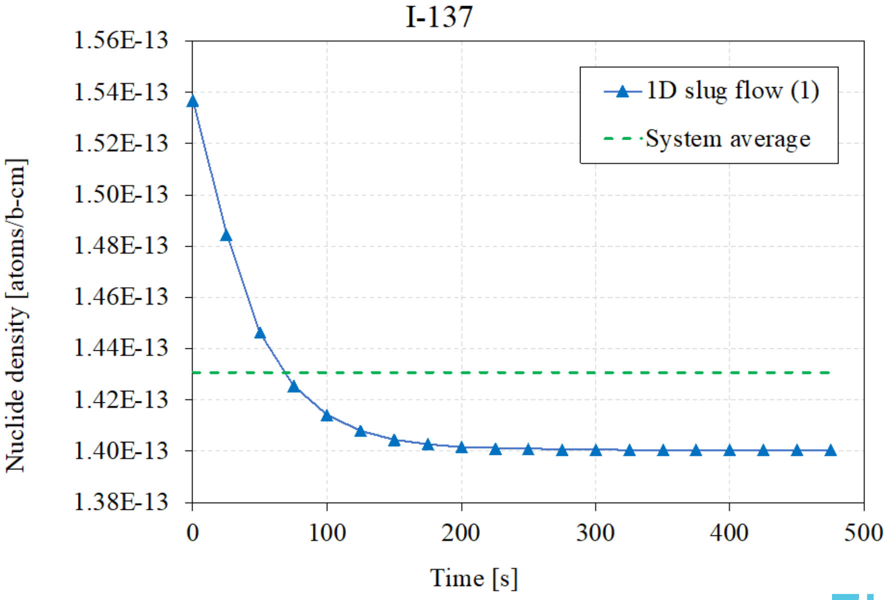


*relative to the loop transit time (~25 s for MSRE)

Short-lived nuclide (I-137, $t_{1/2}$=24.5s) as a function of location in the loop



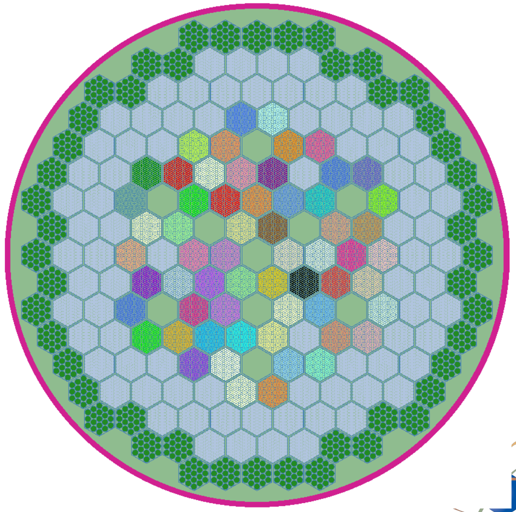
Short-lived nuclide as a function of time at the bottom of the core (zone 1)

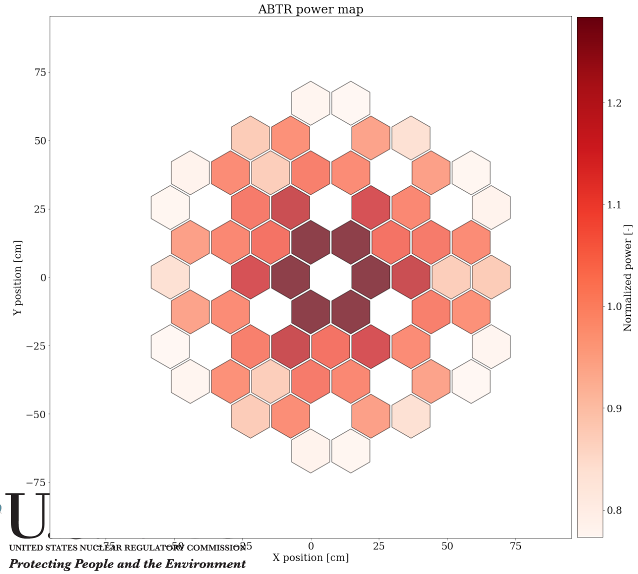


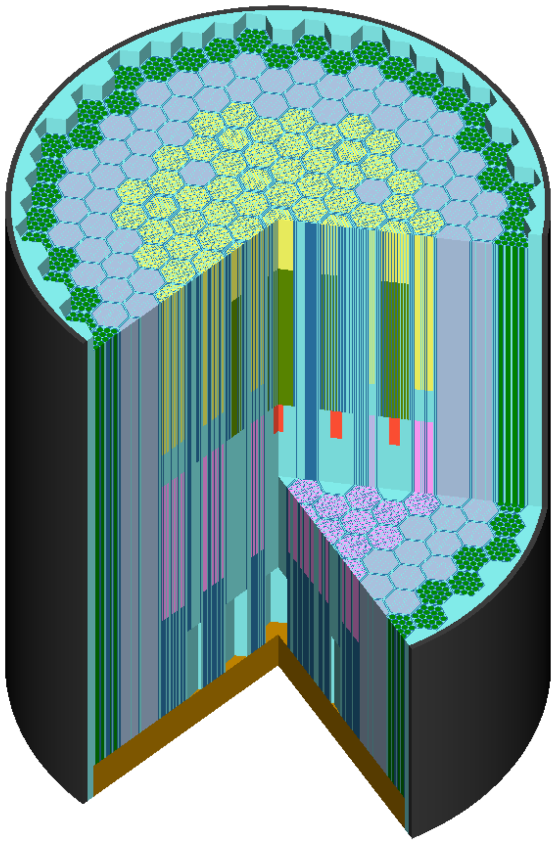
SCALE analysis approach for SFR

- Development of fully heterogeneous full-core model for continuous-energy Monte Carlo calculation
- Power-profile calculation via axial and radial discretization of fuel region
- Full-core depletion calculation to obtain core inventory at end of cycle
- Reactivity effect calculations via direct perturbations: coolant density, fuel temperature, fuel axial expansion, radial core expansion, etc.

ABTR model with individual assembly definitions and corresponding power map

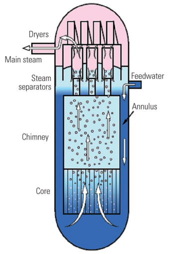


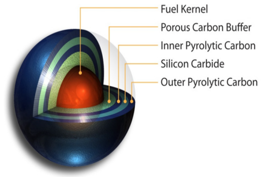




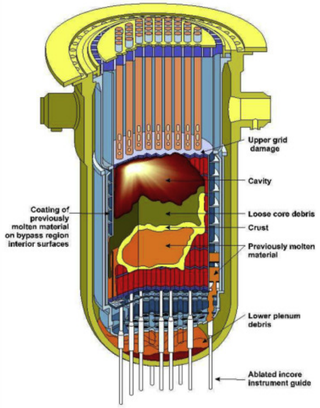
SCALE ABTR model

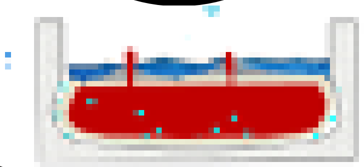
MELCOR Modeling Scope

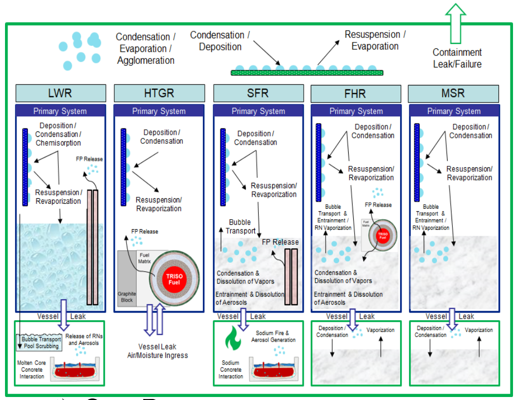




- Thermal hydraulics
- Fuel thermal-mechanical response
- Core degradation
- Ex-vessel damage progression
- Fission product release and transport
- Reactivity Effects







Oak Ridge National Laboratory

NRC Nuclear Regulatory Commission
Protecting People and the Environment

MELCOR Non-LWR Modeling

Hydrodynamic modeling

- Generalized working fluid treatment
- *Conduction heat transfer within working fluids (under development)*
- Generalized convection and flow models to capture flow through new core geometries (e.g., pebble beds)

Core models

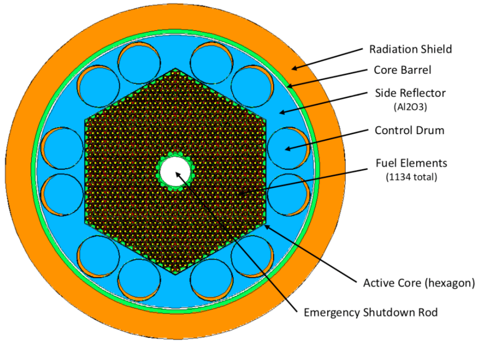
- TRISO pebble and compact core components
- Heat pipe reactor core component
- Graphite oxidation
- Intercell and intracell conduction
- *Fast reactor core degradation (under development)*

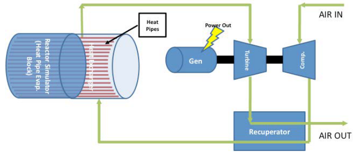
Fission product release

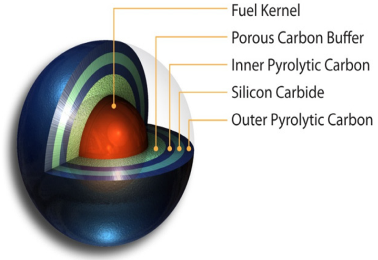
- Generalized release modeling for metallic fuels
- Radionuclide transport and release from TRISO particles, pebbles and compacts
- *Generalized Radionuclide Transport and Retention (GRTR) model (under development)*

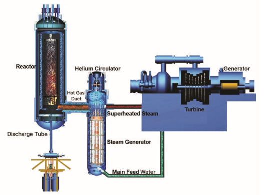
Simplified neutronic modeling

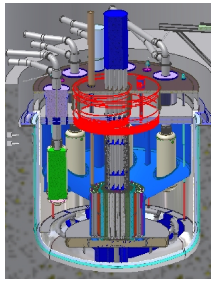
- Solid fuel core point kinetics
- Fluid point kinetics (liquid-fueled molten salt reactors)

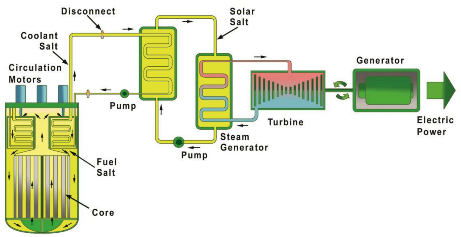


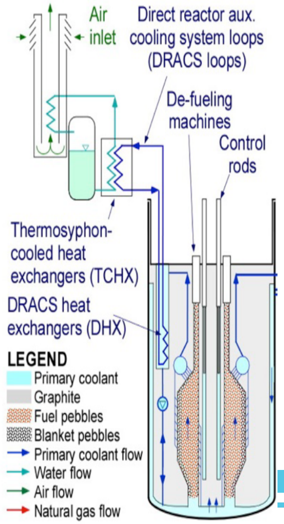






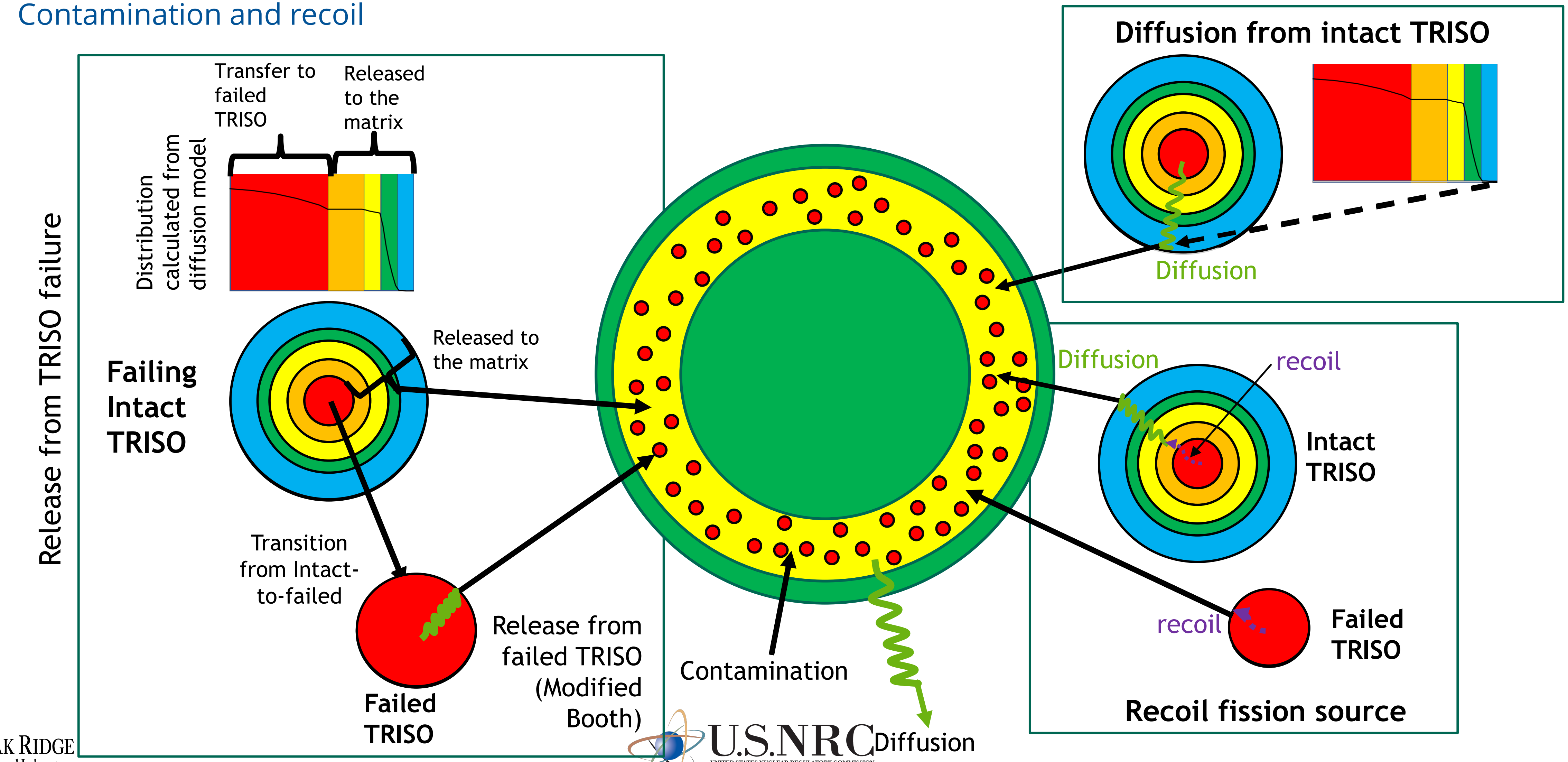






TRISO Radionuclide Release Modeling

Recent failures – particles failing within latest time-step (burst release, diffusion release in time-step)
Previous failures – particles failing on a previous time-step (time history of diffusion release)
Contamination and recoil



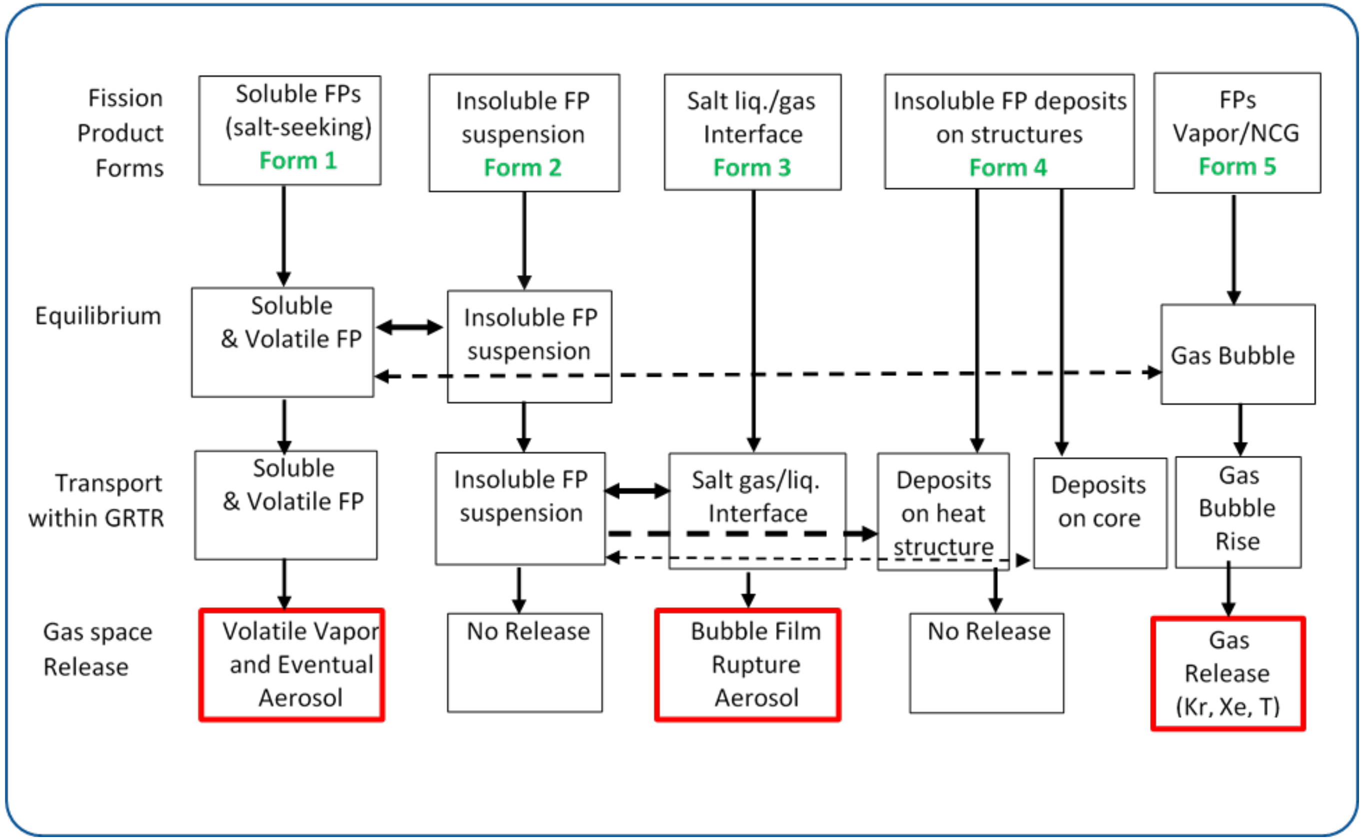
MELCOR Generalized Radionuclide Transport and Retention (GRTR) Model

Model Scope

Uses 5 radionuclide physico-chemical forms in liquid pool
- Soluble fission products
- Insoluble fission products suspended in working fluid
- Insoluble fission products deposited on structures
- Insoluble fission products at liquid-gas interface
- Fission product gases

Generalized Gibbs Energy Minimization approach
- Fission product solubility
- Fission product vapor pressure

Model generically applies to range of non-LWR working fluids
- Molten salt systems
- Liquid metal systems

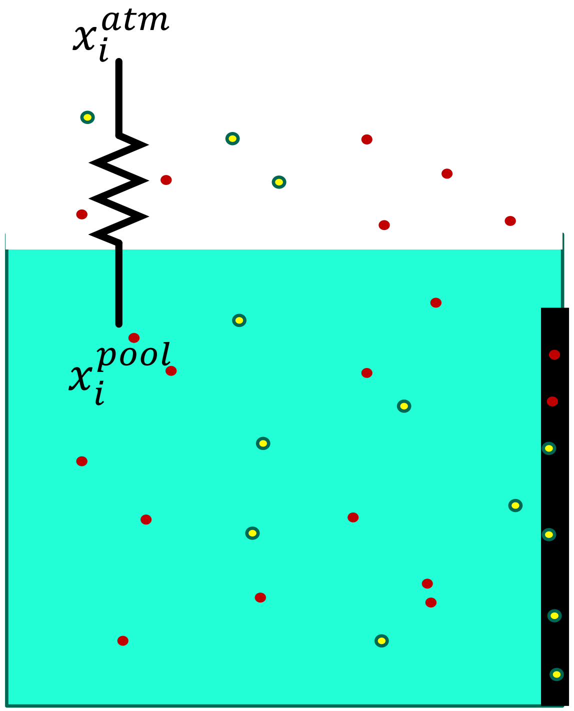


Radionuclides grouped into forms found in the Molten Salt Reactor Experiment

Oak Ridge National Laboratory

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

MELCOR Generalized Radionuclide Transport and Retention (GRTR) – States and State Transitions



Radionuclides characterized in terms of…

Isotopic state

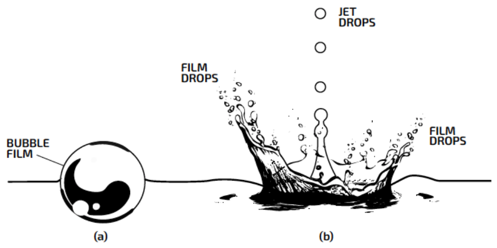
- Fission product decay

Distribution of fission products in reactor system

- Hydrodynamic flows moving fission products within system

Physico-chemical form and ability of fission products to be transported out of the liquid

- Deposition on structures from the liquid
- Vaporization into gas atmospheres from the liquid
- Attachment to gas bubbles
- Aerosolization of fission products into atmosphere above the liquid via bursting of bubbles



Note: MELCOR considers soluble, bulk colloid, interfacial colloid, and vapors as distinct chemical states

Cesium Vaporization from Molten Salt – FHR Example

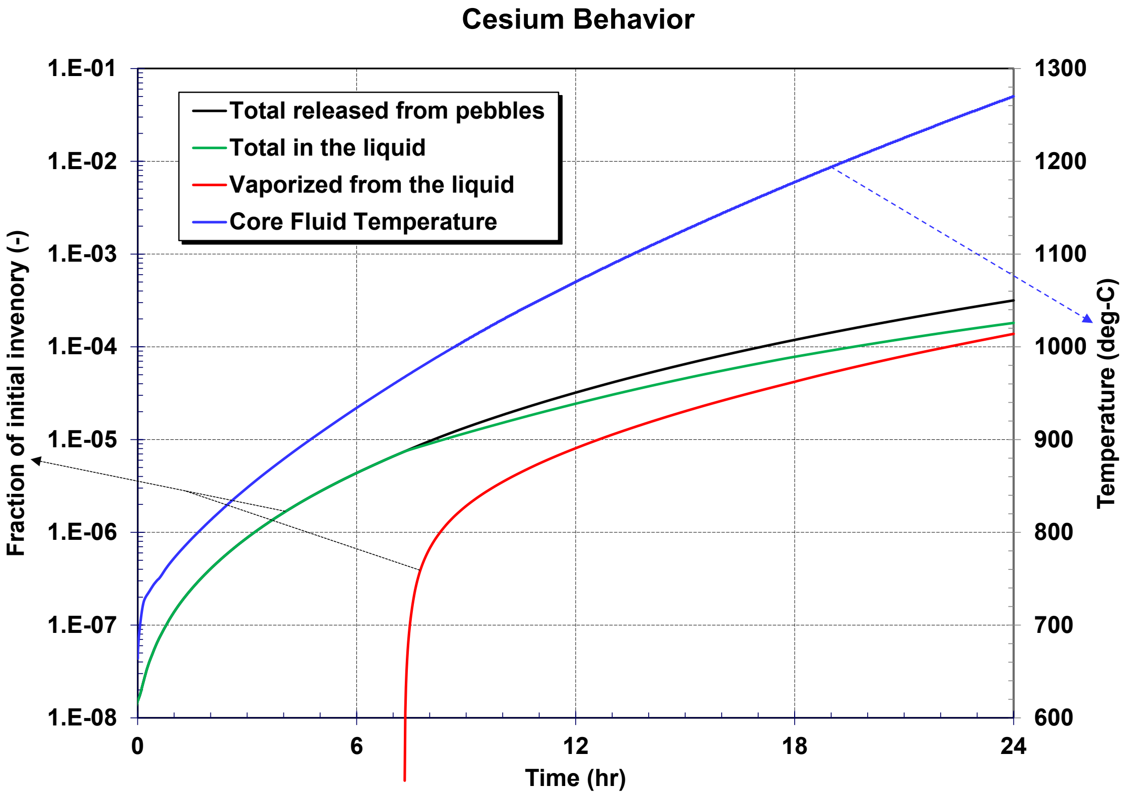
Fission product thermochemistry modeling sample demonstration

- Exercise machinery
- Focuses on Cs and CsF release from salt pool
- Thermochimica Gibbs Energy Minimizer
- Utilizing vapor phase data for CsF*

Demonstration calculation for LOCA sequence

- No core uncovery through 24 hours

Model exhibits Cs and CsF vaporization to gas space at elevated salt temperatures



Cs Transport Pathway

Fuel Pebbles → Molten Salt → Overlying Gas Atmosphere

Oak Ridge National Laboratory

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

* With modifications by Ontario Tech.

Point Kinetics Modeling

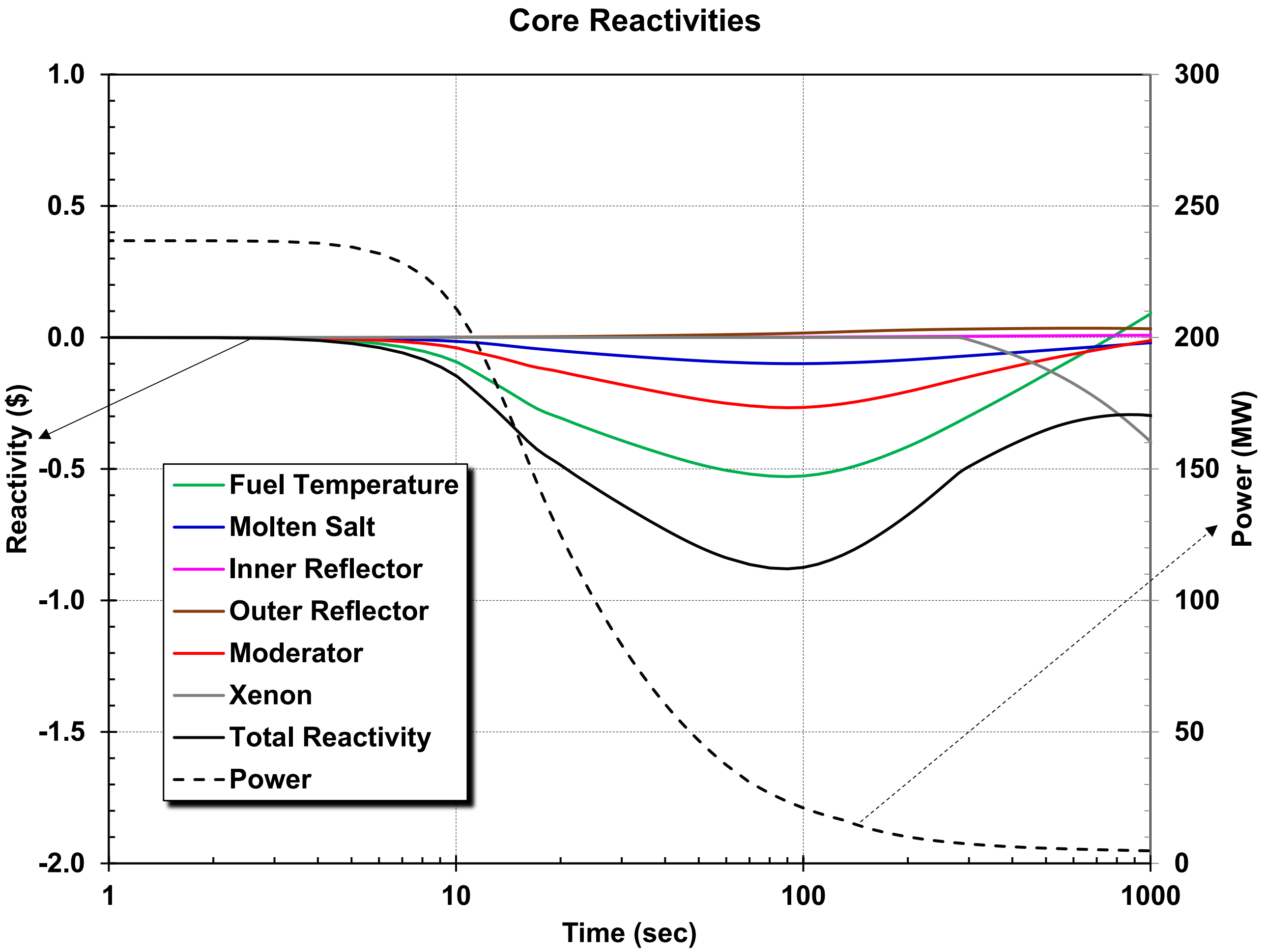
Some accidents may involve reactivity feedbacks

For non-LWRs, MELCOR uses a point kinetics models

Feedback models

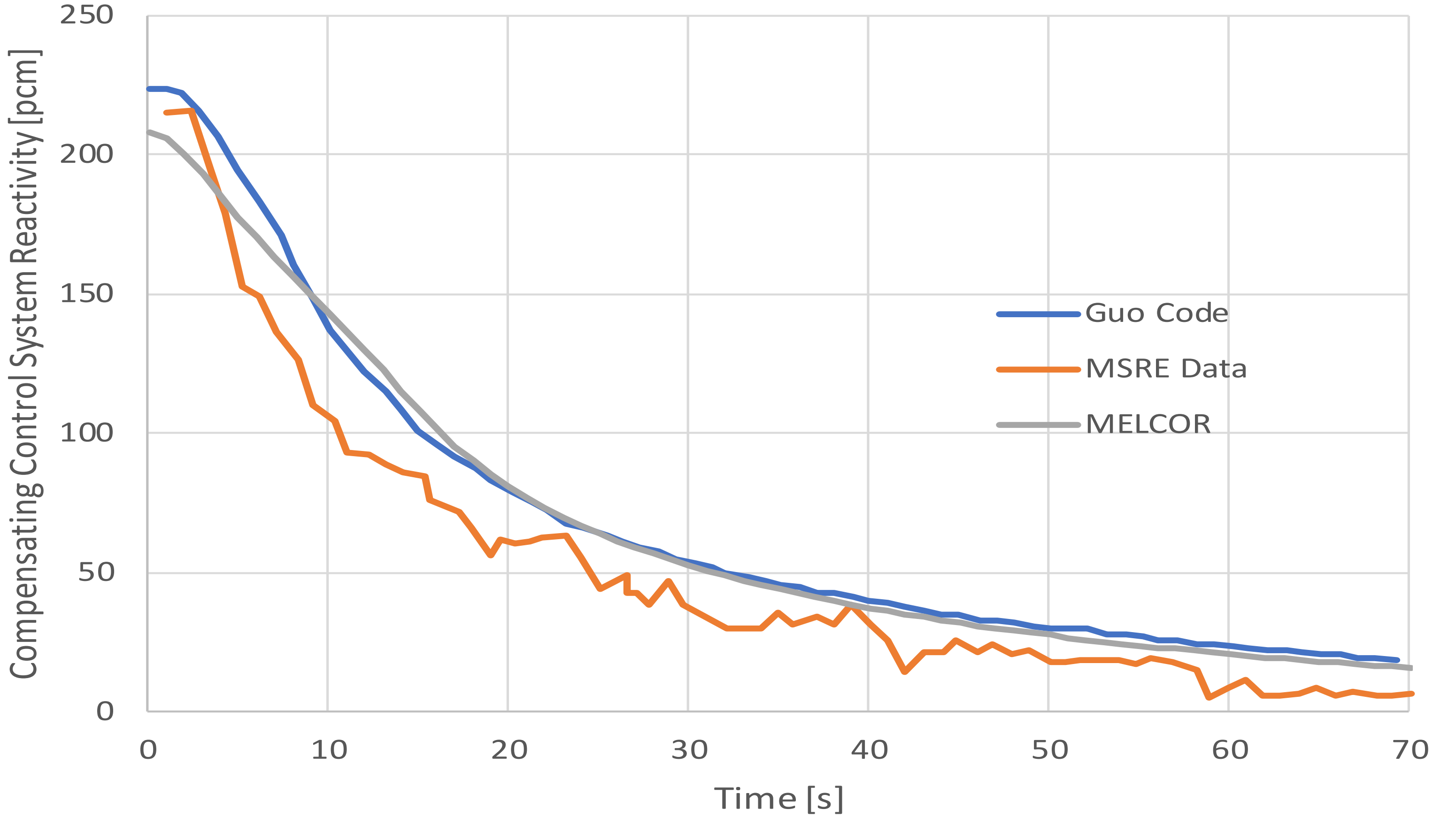
- User-specified external input
- Doppler
- Fuel and moderator density
- Flow reactivity feedback effects integrated into the equation set

FHR example calculation using MELCOR point kinetics model



Point Kinetics Modeling (MSR)

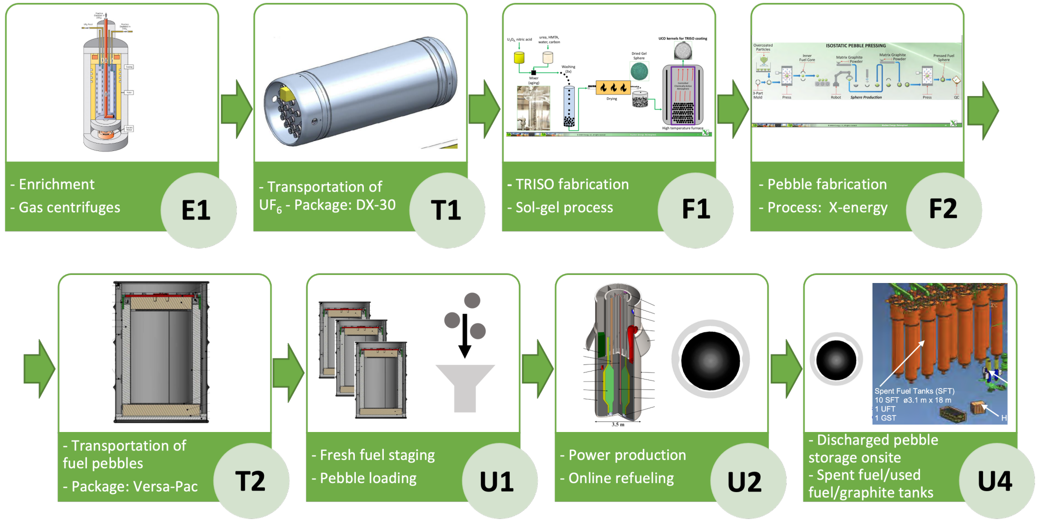
Extended static point kinetic equations to capture motion of delayed precursors through the reactor system



Validated against MSRE zero-power flow experiments

NRC Non-LWR Vision and Strategy, Volume 5 Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle

- Project goal: Demonstration of capabilities to simulate accident scenarios during the fuel cycle with MELCOR and SCALE for HTGR, SFR, MSR, HPR, FHR
- Current effort is the development of the project plan:
  - Determine boundary conditions for each stage of the fuel cycle
  - Identify potential hazards and accident scenarios for each stage of the fuel cycle
  - From these, select accident scenarios for SCALE/MELCOR to simulate
- Challenges encountered:
  - Some stages of the fuel cycle are not yet developed
  - Many documents are proprietary (e.g., safety analysis reports)
- Current status:
  - HTGR fuel cycle developed and discussed between ORNL/SNL/NRC
  - MSR and SFR fuel cycle discussions scheduled for end of January/early February



HTGR fuel cycle

Advanced Reactor Stakeholder Public Meeting

Break

Meeting will resume in 10 minutes

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 323 588 045#

NRC Activities on Advanced Manufacturing Technologies (AMTs)

Matthew Hiser

NRC Office of Nuclear Regulatory Research

January 19, 2022

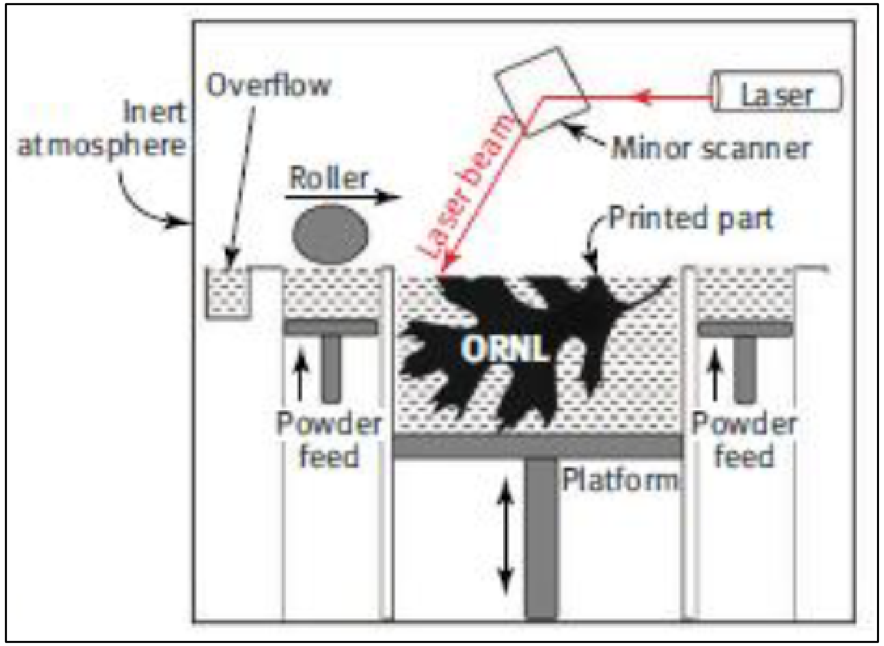
Periodic Advanced Reactor Stakeholder Meeting

Advanced Manufacturing Technologies

- Techniques and material processing methods that have **not** been:
  - Traditionally used in the U.S. nuclear industry
  - Formally standardized/codified by the nuclear industry
- Key AMTs based on industry interest:
  - Laser Powder Bed Fusion (LPBF)
  - Directed Energy Deposition (DED)
  - Electron Beam Welding (EBW)
  - Powder Metallurgy - Hot Isostatic Pressing (PM-HIP)
  - Cold Spray (CS)

Laser Powder Bed Fusion

- Process:
  - Uses laser to melt or fuse powder particles together within a bed of powder
  - Generally most advantageous for more complex geometries



Schematic of LPBF process

- Potential LWR Applications
  - Smaller Class 1, 2 and 3 components, fuel hardware, small internals

First US Application of Additive Manufacturing

- Thimble Plugging Device
  - Installed in March 2020 in Byron Unit 1
  - 316L stainless steel -LPBF
  - Very low safety significant component (Non ASME B&PV Code class)
  - PWR environment with irradiation
  - Installation done without prior NRC approval under 10 CFR 50.59

https://www.neimagazine.com/news/newswestinghouse-produces-3d-printed-component-for-us-nuclear-plant-7911951

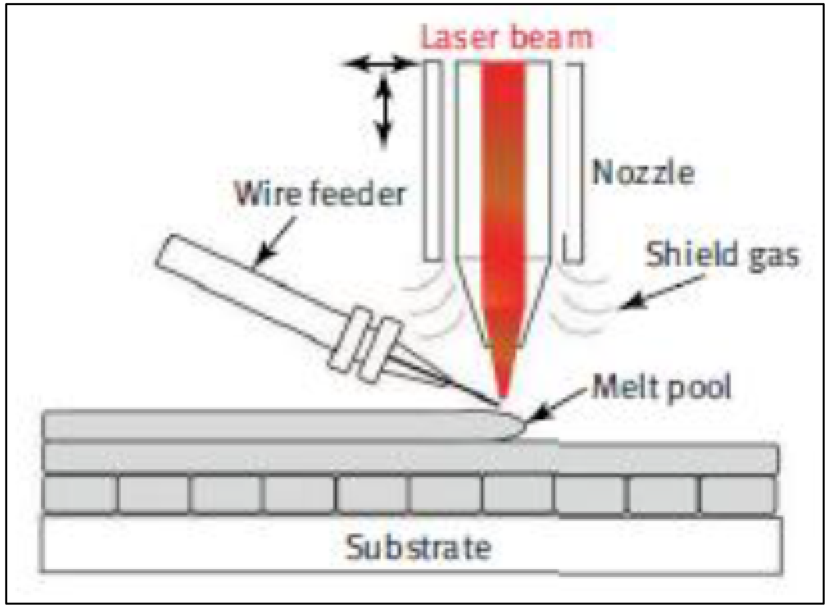
Second US Application of Additive Manufacturing

- Channel Fastener
  - Installed in April 2021 at Browns Ferry Unit 2
  - 316L stainless steel - LPBF
  - Non ASME B&PV Code Class
  - BWR environment with irradiation
  - Installation done without prior NRC approval under 10 CFR 50.59

https://www.ornl.gov/news/additively-manufactured-components-ornl-headed-tva-nuclear-reactor?utm_source=miragenews&utm_medium=miragenews&utm_campaign=news

Directed Energy Deposition

- Process:
  - Wire or powder fed through nozzle into laser or electron beam
  - Fundamentally welding using robotics/ computer controls



Schematic of DED process

- Potential Applications
  - Similar to LPBF, although larger components possible due to faster production and greater build chamber volumes

Powder Metallurgy – Hot Isostatic Pressing (PM-HIP)

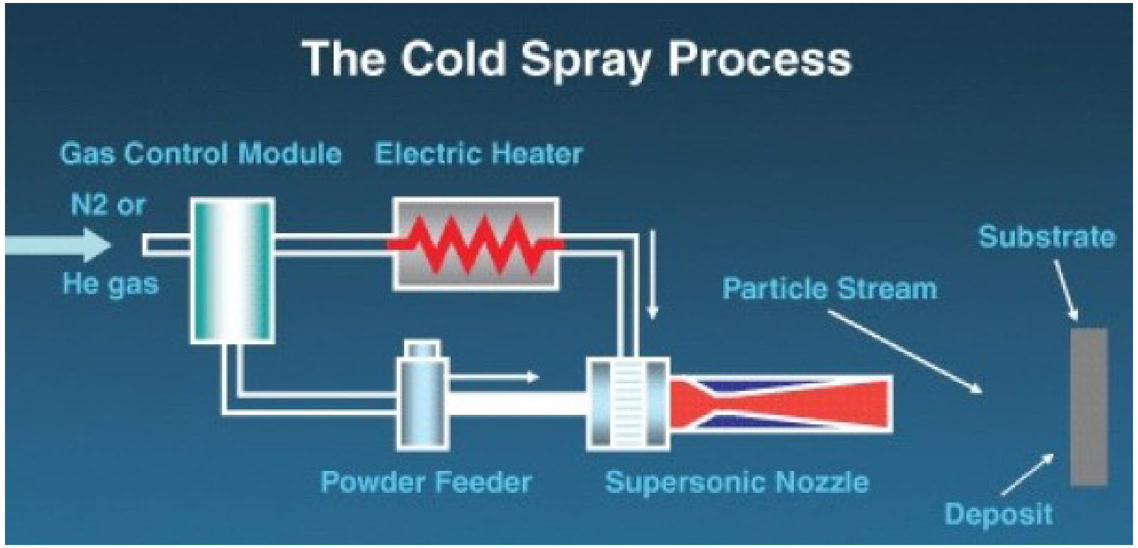
- Process:
  - Metal powder is encapsulated in a form mirroring the desired part
  - The encapsulated powder is exposed to high temperature and pressure, densifying the powder and producing a uniform microstructure
  - After densification, the capsule is removed, yielding a near-net shape component where final machining and inspection can be performed
- Potential Applications
  - All sizes of Class 1, 2 and 3 components and reactor internals
  - EPRI / DOE focused on use with electron beam welding to fabricate NuScale reactor vessel

Electron Beam Welding

- Process:
  - Fusion welding process that uses a beam of high-velocity electrons to join materials
  - Single pass welding without filler metal
  - Welding process can be completed much more quickly due to deep penetration
- Potential Applications
  - For welding medium and large components, such as NuScale upper head

Cold Spray

- Process:
  - Powder is sprayed at supersonic velocities onto a metal surface and forms a bond with the part
  - This can be used to repair existing parts or as a mitigation process



Schematic of cold spray process*

- Potential Applications
  - Mitigation or repair of potential chloride-induced stress corrosion cracking (CISCC) in spent fuel canisters
  - Mitigation or repair of stress corrosion cracking (SCC) in reactor applications

https://www.army.mil/article/148465/army_researchers_develop_cold_spray_system_transition_to_industry

Industry and Research Activities

- Variety of stakeholders are working towards more widespread use in both existing and future nuclear applications
  - Vendors and licensees/applicants
    - Identifying candidate applications
    - Developing technical basis for gaining regulatory acceptance
  - Nuclear Energy Institute – Developed roadmap to understand industry needs/interests and assist with regulatory acceptance
  - Electric Power Research Institute – Developing techniques for large components in small modular reactors, developed data package for 316L L-PBF ASME draft Code case
  - US Department of Energy – Performing basic and applied research and technology development to support AMT implementation

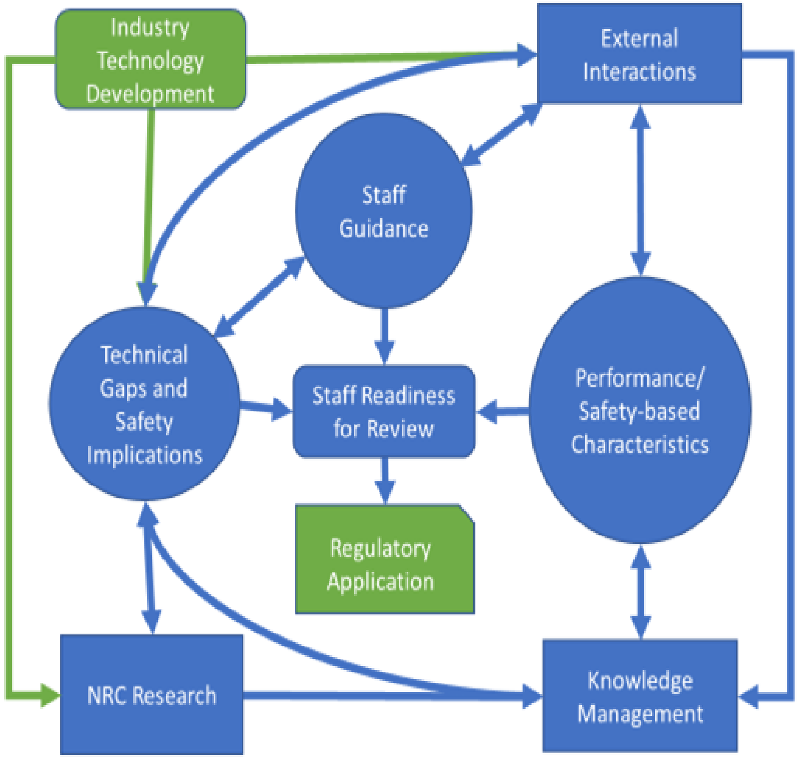
Codes and Standards

- Codes and Standards Organizations (eg ASTM, ASME) – addressing standardization gaps, Code Cases (PM-HIP, LPBF)
  - ASME Special Working Group –
    - Developing guidelines for use of additive manufacturing (AM), “Criteria for Pressure Retaining Metallic Components Using Additive Manufacturing”. Was published as an ASME Pressure Technology Book
    - 316L L-PBF Data Package and Code Case under development
  - ASME Task Group on AM for High Temperature Applications
    - Developing Code actions for incorporating AM materials/components in ASME Section III, Division 5 (high temperature reactors) for elevated temperature nuclear construction
  - ASME PM-HIP Code Case approved for use by US NRC
    - Code Case N-834 allows use of ASTM A988/A988M “Standard Specification for Hot Isostatically-Pressed Stainless Steel Flanges, Fittings, Valves, and Parts for High Temperature Service” in Section III, Division 1 Class 1 components
    - October 2019 - RG 1.84, Revision 38 approved this Code Case as acceptable for use without conditions

NRC Action Plan

- NRC activities related to AMTs have been organized and planned through the AMT action plan with the following objectives:
  - Assess the safety significant differences between AMTs and traditional manufacturing processes, from a performance-based perspective.
  - Prepare the NRC staff to address industry implementation of AMT-fabricated components through the 10 CFR 50.59 process.
  - Identify and address AMT characteristics pertinent to safety, from a risk-informed and performance-based perspective, that are not managed or addressed by codes, standards, regulations, etc.
  - Provide guidance and tools for review consistency, communication, and knowledge management for the efforts associated with AMT reviews.
  - Provide transparency to stakeholders on the process for AMT approvals.
- Revision 1 was published in June 2020 (ML19333B980)

Action Plan – Rev. 1 Tasks



- Task 1 - Technical Preparedness
  - Technical information, knowledge and tools to prepare NRC staff to review AMT applications
- Task 2 - Regulatory Preparedness
  - Regulatory guidance and tools to prepare staff for efficient and effective review of AMT-fabricated components submitted to the NRC for review and approval
- Task 3 - Communications and Knowledge Management
  - Integration of information from external organizations into the NRC staff knowledge base for informed regulatory decision-making
  - External interactions and knowledge sharing, i.e. AMT Workshop (held in Dec. 2020)

Task 1 Technical Preparedness Activities

- Subtask 1A: AMT Processes under Consideration
  - Perform a technical assessment of multiple selected AMTs of interest
  - Gap assessment for each selected AMTs vs traditional manufacturing techniques
  - Technical letter report and technical assessment for each AMT: LPBF - ML20351A292
- Subtask 1B: NDE Gap Assessment
  - Literature survey of the current state of the art of non-destructive examination (NDE) of components made using advanced manufactured technologies (AMTs) (ML20349A012).
- Subtask 1C: Microstructural and Modeling
  - Evaluate modeling and simulation tools used to predict the initial microstructure, material properties and component integrity of AMT components
  - Identify existing gaps and challenges that are unique to AMT compared to conventional manufacturing processes:
    - Predicting Initial Microstructures (ML20269A301); Predicting Material Performance (ML20350B550)

Task 2 - Regulatory Preparedness Activities

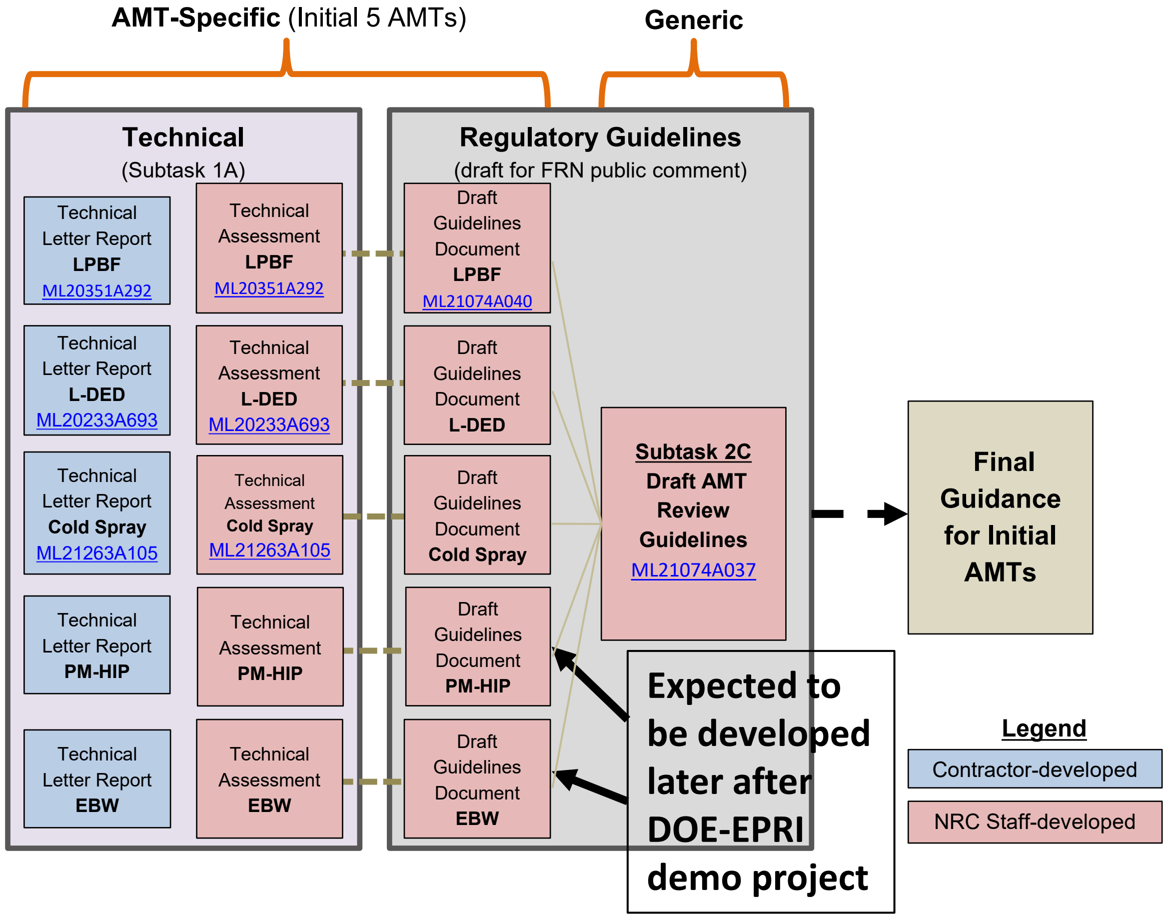
- Subtask 2A: Implementation using the 10 CFR 50.59 Process
  - Provide guidance and support to regional inspectors regarding AMTs implemented under quality assurance and 50.59 programs. Complete: ML21155A043

- Subtask 2B: Assessment of Regulatory Guidance
  - Assess whether any regulatory guidance needs to be updated or created to clarify the process for reviewing submittals with AMT components. Complete: ML20233A693

- Subtask 2C: AMT Guidelines Document
  - Develop a report which describes the generic technical information to be addressed in AMT submissions. Technology specific guidelines are also being developed.
  - Public meeting held on September 16, 2021 to discuss Draft AMT Review Guidelines ML21074A037 and Draft Guidelines Document for AM –LPBF ML21074A040

NRC AMT Guidelines Development

- A Technical Letter Report (TLR) is produced for each of the initial five AMTs
  - Provides technical basis information and gap analysis
  - Written by NRC contractor (to date, DOE labs)
- A technical assessment (TA) is produced for each TLR by NRC staff which provides the NRC staff perspective on key aspects of the AMT for safety and component performance
- A draft guidelines document (DGD), informed by the TA and TLR, will be generated by the NRC staff for each AMT.
  - The AMT-specific DGDs accompany and align with the generic Advanced Manufacturing Technologies Review Guidelines



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

Communications and KM Activities

- Subtask 3A: Internal Interactions
  - Internal coordination with NRC staff in other areas (e.g., advanced reactors, dry storage, fuels)
- Subtask 3B: External Interactions
  - Engagement with codes and standards, industry, research, international
- Subtask 3C: Knowledge Management
  - Seminars, public meetings, training, knowledge capture tools
- Subtask 3D: Public Workshop
  - RIL 2021-03: Part 1 Part 2
- Subtask 3E: AMT Materials Information Course
  - Internal NRC staff training
  - Six seminars to date on a variety of topics

Status of Deliverables – Task 1

Subtask	Actions/Deliverables	Status
1A AMT processes under consideration	Additive Manufacturing (AM) – Laser Powder Bed Fusion	Complete - ML20351A292
	AM – Directed Energy Deposition (DED)	Complete - ML20233A693
	Cold Spray	Complete - ML21263A105
	Powder Metallurgy (PM) – Hot Isostatic Pressing (HIP)	Draft report under NRC review
	Electron Beam (EB) welding	Draft report under NRC review
1B Inspection and NDE	PNNL NDE gap analysis	Complete - ML20349A012
1C Modeling and Simulation of Microstructure	ANL M&S gap analysis to predict microstructure	Complete - ML20269A301
	ANL M&S gap analysis to predict material performance	Complete - ML20350B550

Status of Deliverables – Tasks 2 and 3

Subtask	Actions / Deliverables	Status
2A 50.59 process	Finalize document incorporating feedback from Regional staff regarding the 10 CFR 50.59 process	Complete – ML21200A222
2B Assessment of regulatory guidance	Path forward on guidance development or modification	Complete - ML20233A693
2C AMT Guidance Document	Public meeting on guidance concept / framework	Public meeting held on July 30, 2020 – summary: ML20240A077
	Develop a document that describes the generic technical information to be addressed in AMT submittals.	Public meeting held on September 16, 2021 to discuss: ML21074A037 - Draft AMT Review Guidelines ML21074A040 - Draft Guidelines Document for AM – LPBF
	Public meeting to discuss draft document	
3A/3B External/ Internal Interactions	Continued communication with NRC staff, vendors, licensees and EPRI for future AMTs	Ongoing as needed
3C Knowledge Management Plan	Develop Knowledge Management Plan	Complete – internal
3D Workshop	Hold Public Workshop	Complete – summary: ML20357B071 RIL: Part 1 Part 2
3E Material Information course	Training course and course materials	First 6 seminars complete – internal

Path Forward

- Complete remaining activities under Rev. 1 AMT Action Plan:
  - EBW and PM-HIP technical report and assessment
  - L-DED and Cold spray DGDs
- Plan and initiate future work likely focused on:
  - Additional AMTs
  - In-process NDE and digital data for qualification
  - AMT guidance development
  - Knowledge management and staff training on AMTs

Future Meeting Planning

- The next periodic stakeholder meeting is scheduled for March 16, 2022.
- If you have suggested topics, please reach out to Prosanta.Chowdhury@nrc.gov.