Advanced Reactor Stakeholder Public Meeting

January 19, 2022

Microsoft Teams Meeting
Bridgeline: 301-576-2978
Conference ID: 323 588 045#
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
<table>
<thead>
<tr>
<th>Time</th>
<th>Agenda</th>
<th>Speaker</th>
</tr>
</thead>
<tbody>
<tr>
<td>10:00 – 10:20 am</td>
<td>Opening Remarks / Adv. Rx Integrated Schedule</td>
<td>NRC</td>
</tr>
<tr>
<td>10:20 – 10:30 am</td>
<td>Status Overview of the Adv. Rx Generic Environmental Impact Statement (GEIS) and Rulemaking Activities</td>
<td>NRC</td>
</tr>
<tr>
<td>10:30 – 11:15 am</td>
<td>Implementing Near-field Models in MACCS v4.1 for Better Near-field Dose Calculations</td>
<td>NRC/SNL</td>
</tr>
<tr>
<td>11:15 am – 12:00 pm</td>
<td>Light Water Reactor Construction Permit Interim Staff Guidance</td>
<td>NRC</td>
</tr>
<tr>
<td>12:00 – 1:00 pm</td>
<td>Lunch Break</td>
<td>All</td>
</tr>
<tr>
<td>1:00 – 1:45 pm</td>
<td>Nuclear Data Assessment for Advanced Reactors</td>
<td>NRC/ORNL</td>
</tr>
<tr>
<td>1:45 – 2:30 pm</td>
<td>SCALE/MELCOR Development and Applications for non-LWRs</td>
<td>NRC/SNL &amp; ORNL</td>
</tr>
<tr>
<td>2:30 – 2:40 pm</td>
<td>Break</td>
<td>All</td>
</tr>
<tr>
<td>2:40 – 3:20 pm</td>
<td>Advanced Manufacturing Technologies</td>
<td>NRC</td>
</tr>
<tr>
<td>3:20 – 3:30 pm</td>
<td>Future Meeting Planning and Concluding Remarks</td>
<td>NRC</td>
</tr>
</tbody>
</table>
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 Program - Summary of Integrated Schedule and Regulatory Activities*

<table>
<thead>
<tr>
<th>Strategy</th>
<th>Regulatory Activity</th>
<th>2021</th>
<th>2022</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Development of non-Light Water Reactor (LWR) Training for Advanced Reactors (Adv. Rxs) (NEIMA Section 103(a)(5))</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>2</td>
<td>FAST Reactor Technology</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>3</td>
<td>Molten Salt Reactor (MSR) Technology</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>4</td>
<td>Code Assessment Reports Volume 1 (Systems Analysis)</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>5</td>
<td>Reference SCALE/MELCOR plant model for Heat Pipe-Cooled Micro Reactor (update from v1 to v2)</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>6</td>
<td>Reference SCALE/MELCOR plant model for Heat Pipe-Cooled Pebble Bed Reactor (update from v1 to v2)</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>7</td>
<td>Code Assessment Reports Volume 2 (Fuel Perfr. Analysis)</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>8</td>
<td>Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>9</td>
<td>Code Assessment Reports Volume 3 (Source Term Analysis)</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>10</td>
<td>Reference SCALE/MELCOR plant model for Sodium-Cooled Fast Reactor</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>11</td>
<td>Code Assessment Reports Volume 4 (Licensing and Siting Dose Assessments)</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>12</td>
<td>Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>13</td>
<td>Code Assessment Reports Volume 5 (Fuel Cycle Analysis)</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>14</td>
<td>Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
<tr>
<td>15</td>
<td>Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor</td>
<td>Present Day</td>
<td>Present Day</td>
</tr>
</tbody>
</table>

*Legend: **Legend:**

- **Red:** NEIMA
- **Yellow:** EDO
- **Green:** SCCFC
- **Blue:** Commission Review Period
- **Purple:** ACRS SC/CFC

---

**Note:**
- **Strategy 1:** Development of non-Light Water Reactor (LWR) Training for Advanced Reactors (Adv. Rxs) (NEIMA Section 103(a)(5))
- **Strategy 2:** FAST Reactor Technology
- **Strategy 3:** Molten Salt Reactor (MSR) Technology
- **Strategy 4:** Code Assessment Reports Volume 1 (Systems Analysis)
- **Strategy 5:** Reference SCALE/MELCOR plant model for Heat Pipe-Cooled Micro Reactor (update from v1 to v2)
- **Strategy 6:** Reference SCALE/MELCOR plant model for Heat Pipe-Cooled Pebble Bed Reactor (update from v1 to v2)
- **Strategy 7:** Reference SCALE/MELCOR plant model for Sodium-Cooled Fast Reactor
- **Strategy 8:** Reference SCALE/MELCOR plant model for Sodium-Cooled Fast Reactor
- **Strategy 9:** Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor
- **Strategy 10:** Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor
- **Strategy 11:** Reference SCALE/MELCOR plant model for Sodium-Cooled Fast Reactor
- **Strategy 12:** Reference SCALE/MELCOR plant model for Sodium-Cooled Fast Reactor
- **Strategy 13:** Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor
- **Strategy 14:** Reference SCALE/MELCOR plant model for Molten Salt Fueled Reactor
- **Strategy 15:** Reference SCALE/MELCOR plant model for Sodium-Cooled Fast Reactor

---

**References:**
- [Reference plant model for Sodium-Cooled Fast Reactor](https://www.nrc.gov/reading-rm/doc-collections/federal-register/)
- [MACCS radionuclide screening analysis](https://www.nrc.gov/reading-rm/doc-collections/federal-register/)
- [MACCS near-field atmospheric transport and dispersion model improvement](https://www.nrc.gov/reading-rm/doc-collections/federal-register/)
- [Phase 1 – Atmospheric Code Consolidation](https://www.nrc.gov/reading-rm/doc-collections/federal-register/)
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
- “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

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

https://www.nrc.gov/reactors/new-reactors/advanced/details#advSumISRA
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
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)
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](https://www.nrc.gov/reading-rm/adams.html#web-based-adams)

<table>
<thead>
<tr>
<th>Document</th>
<th>ADAMS Accession No.</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Proposed Rule Package: SECY-21-0098:</strong> Proposed Rule: Advanced Nuclear Reactor Generic Environmental Impact Statement (RIB3150-AK55; NRC-2020-0101)</td>
<td>ML21222A044</td>
</tr>
<tr>
<td><strong>Preliminary Draft Guide-4032 Package:</strong> Preliminary Draft Guide-4032 (RG 4.2), “Preparation of Environmental Reports for Nuclear Power Stations”</td>
<td>ML21208A111</td>
</tr>
<tr>
<td><strong>Preliminary Draft of Interim Staff Guidance COL-ISG-30:</strong> 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)</td>
<td>ML21227A005</td>
</tr>
</tbody>
</table>
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

Dan Clayton
MACCS Principal Investigator
Sandia National Laboratories

Advanced Reactor Stakeholder Meeting
January 19, 2022
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 \textit{gaussian plume equation} with reflective boundaries and includes \textit{models} for \textit{plume meander} and \textit{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 \textit{simple model} for building wake effects. The MACCS User’s Guide suggests that this simple building wake model \textbf{should not be used at distances closer than 500 m}. This statement raised the question of \textbf{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)

<table>
<thead>
<tr>
<th>Model</th>
<th>Simplicity</th>
<th>Efficiency</th>
<th>Validation</th>
<th>Conservative Bias</th>
<th>Community Acceptance</th>
<th>Ease of Implementation</th>
</tr>
</thead>
<tbody>
<tr>
<td>OpenFOAM</td>
<td>3</td>
<td>3</td>
<td>1</td>
<td>2</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>QUIC</td>
<td>3</td>
<td>2</td>
<td>1</td>
<td>2</td>
<td>2</td>
<td>3</td>
</tr>
<tr>
<td>ARCON96</td>
<td>1</td>
<td>1</td>
<td>2</td>
<td>2</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>AERMOD</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>2</td>
<td>1</td>
<td>2</td>
</tr>
</tbody>
</table>
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 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

\[ \Sigma_y = (\sigma_y^2 + \Delta\sigma_y^2 + \Delta\sigma^2)^{1/2} \]

\[ \Sigma_z = (\sigma_z^2 + \Delta\sigma_z^2 + \Delta\sigma^2)^{1/2} \]

Plume Meander

- US NRC Regulatory Guide 1.145 (MNDMOD=NEW)
- Ramsdell and Fosmire (MNDMOD=RAF)
- Original MACCS (MNDMOD=OLD)
- None (MNDMOD = OFF)

Reg. Guide 1.145

\[ \chi/Q = \frac{1}{U_{10}(\pi\sigma_y\sigma_z + A/2)} \] (1)

\[ \chi/Q = \frac{1}{U_{10}(3\pi\sigma_y\sigma_z)} \] (2)

\[ \chi/Q = \frac{1}{U_{10}\pi\Sigma_y\sigma_z} \] (3)
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
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.
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.

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](https://maccs.sandia.gov) for more information)
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
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](https://www.federalregister.gov/articles/2021/12/15/2021-25923/nuclear-regulatory-commission))

The draft interim staff guidance may be found in the NRC’s Agencywide Documents Access and Management System at this link: [ML21165A157](https://www.nrc.gov/public- podrá)
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/](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](mailto: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](mailto: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#
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
Motivation/Background
Commercial Light Water Reactor Approach to Reactor Physics/Nuclear Design

- 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

- 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

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)

Emphasized during safety review
Impact of Data Uncertainty

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

• Advanced Reactor examples*:
  – Changes in graphite data from ENDF/B-VII.0 to B-VII.1 (capture cross section) had a 1% $\Delta k/k$ impact
  – No data for FLiBe/FLiNak thermal scattering, possible 2% $\Delta k/k$ 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?
Overview of NUREG/CR-7289, “Nuclear Data Assessment for Advanced Reactors”
Technologies Considered

- High Temperature Gas Reactor
- Fluoride Salt-Cooled High Temperature Reactor
- Graphite Moderated Molten Salt Reactor
- Molten Chloride Fast Spectrum Reactor
- Heat Pipe Microreactor
- Sodium-Cooled Fast Reactor
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

<table>
<thead>
<tr>
<th>Reactor technology</th>
<th>Selected benchmark(^a)</th>
<th>Type</th>
</tr>
</thead>
<tbody>
<tr>
<td>High Temperature Gas Reactor</td>
<td>HTR-10</td>
<td>Experiment</td>
</tr>
<tr>
<td>Fluoride Salt Cooled High Temperature Reactor</td>
<td>UC Berkeley Mark1 PB-FHR</td>
<td>Computational benchmark</td>
</tr>
<tr>
<td>Graphite-moderated Molten Salt Reactor</td>
<td>MSRE</td>
<td>Experiment</td>
</tr>
<tr>
<td>Heat Pipe Microreactor (metal-fueled)</td>
<td>INL Megapower Design A(^b)</td>
<td>Computational benchmark</td>
</tr>
<tr>
<td>Sodium Cooled Fast Reactor (metal and oxide fueled)</td>
<td>EBR-II</td>
<td>Experiment</td>
</tr>
<tr>
<td></td>
<td>ABR-1000</td>
<td>Computational benchmark</td>
</tr>
</tbody>
</table>

\(^a\) 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.
Analyses were performed using ENDF/B-VII.0, ENDF/B-VII.1, and ENDF/B-VIII.0.

Sensitivity coefficients:

\[ S_{Y,\Sigma^i_{x,g}} = \frac{\Sigma^i_{x,g}}{Y} \frac{dY}{d\Sigma^i_{x,g}} \] ; (\( Y \) is the QOI, and \( \Sigma^i_{x,g} \) 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

<table>
<thead>
<tr>
<th>QOIs</th>
<th>ENDF/B-VII.0</th>
<th>ENDF/B-VII.1</th>
<th>ENDF/B-VIII.0</th>
<th>VII.1/VII.0 - 1</th>
<th>VIII.0/VII.1 - 1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel temperature</td>
<td>-243 ± 22</td>
<td>-241 ± 25</td>
<td>-222 ± 25</td>
<td>3 ± 33</td>
<td>19 ± 36</td>
</tr>
<tr>
<td>Pebble gr. density</td>
<td>1182 ± 23</td>
<td>1175 ± 23</td>
<td>1201 ± 27</td>
<td>-8 ± 32</td>
<td>26 ± 35</td>
</tr>
<tr>
<td>Pebble gr. impurities</td>
<td>-602 ± 23</td>
<td>-623 ± 23</td>
<td>-588 ± 25</td>
<td>-21 ± 32</td>
<td>35 ± 34</td>
</tr>
<tr>
<td>Structural gr. density</td>
<td>546 ± 25</td>
<td>504 ± 22</td>
<td>543 ± 24</td>
<td>-43 ± 33</td>
<td>40 ± 32</td>
</tr>
<tr>
<td>Structural gr. impurities</td>
<td>-3947 ± 26</td>
<td>-3877 ± 25</td>
<td>-3807 ± 25</td>
<td>70 ± 36</td>
<td>70 ± 35</td>
</tr>
<tr>
<td>Structural gr. temperature</td>
<td>780 ± 24</td>
<td>783 ± 22</td>
<td>798 ± 24</td>
<td>4 ± 33</td>
<td>14 ± 33</td>
</tr>
</tbody>
</table>
# Results and Key Nuclear Data

(Subset of results from HTR-10 benchmark)

## Sensitivity Analysis Results

### Key Nuclear Data Impacting Pebble Graphite Temperature Feedback

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Reaction</th>
<th>Sensitivity (reducing negative $\Delta \rho$)</th>
<th>Nuclide</th>
<th>Reaction</th>
<th>Sensitivity (increasing negative $\Delta \rho$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>u-235</td>
<td>fission</td>
<td>1.196e+00 ± 6.070e-03</td>
<td>b-10</td>
<td>n,(\alpha)</td>
<td>-9.273e-02 ± 1.440e-03</td>
</tr>
<tr>
<td>u-235</td>
<td>(\bar{\nu})</td>
<td>9.976e-01 ± 6.552e-04</td>
<td>u-238</td>
<td>n,(\gamma)</td>
<td>-3.655e-02 ± 1.764e-03</td>
</tr>
<tr>
<td>s-28</td>
<td>elastic</td>
<td>9.796e-03 ± 6.801e-03</td>
<td>n-14</td>
<td>n,(p)</td>
<td>-5.147e-03 ± 1.908e-04</td>
</tr>
<tr>
<td>c</td>
<td>elastic</td>
<td>9.083e-03 ± 9.656e-03</td>
<td>u-235</td>
<td>elastic</td>
<td>-3.560e-03 ± 3.272e-03</td>
</tr>
<tr>
<td>u-238</td>
<td>elastic</td>
<td>8.487e-03 ± 9.148e-03</td>
<td>si-28</td>
<td>n,(\gamma)</td>
<td>-4.577e-04 ± 2.769e-05</td>
</tr>
<tr>
<td>o-16</td>
<td>elastic</td>
<td>6.737e-03 ± 8.590e-03</td>
<td>graphite</td>
<td>n,(\alpha)</td>
<td>-8.149e-04 ± 2.176e-04</td>
</tr>
<tr>
<td>u-235</td>
<td>n,(\gamma)</td>
<td>6.585e-03 ± 1.145e-03</td>
<td>si-28</td>
<td>n,(n')</td>
<td>-3.930e-04 ± 4.912e-04</td>
</tr>
<tr>
<td>n-14</td>
<td>elastic</td>
<td>6.281e-03 ± 6.051e-03</td>
<td>n-14</td>
<td>n,(\gamma)</td>
<td>-2.084e-04 ± 7.821e-06</td>
</tr>
<tr>
<td>graphite</td>
<td>n,(n')</td>
<td>4.702e-03 ± 2.311e-03</td>
<td>ar-40</td>
<td>elastic</td>
<td>-1.988e-04 ± 1.457e-04</td>
</tr>
<tr>
<td>u-238</td>
<td>nu-fission</td>
<td>2.402e-03 ± 6.552e-04</td>
<td>n-14</td>
<td>n,(\alpha)</td>
<td>-4.236e-05 ± 1.867e-06</td>
</tr>
</tbody>
</table>
Results and Key Nuclear Data
(Subset of results from HTR-10 benchmark)
Uncertainty Analysis Results

Uncertainty in QOIs due to nuclear data

<table>
<thead>
<tr>
<th>QOIs</th>
<th>ENDF/B-VII.0</th>
<th>ENDF/B-VII.1</th>
<th>ENDF/B-VIII.0</th>
<th>VII.1/VI.0</th>
<th>VII.0/VI.1</th>
</tr>
</thead>
<tbody>
<tr>
<td>$k_{\text{eff}}$</td>
<td>0.607%</td>
<td>0.668%</td>
<td>0.690%</td>
<td>10.1%</td>
<td>3.3%</td>
</tr>
<tr>
<td>Fuel temperature</td>
<td>1.124%</td>
<td>1.192%</td>
<td>1.030%</td>
<td>6.1%</td>
<td>-13.6%</td>
</tr>
<tr>
<td>Pebble gr. density</td>
<td>0.667%</td>
<td>0.848%</td>
<td>0.618%</td>
<td>27.1%</td>
<td>-27.1%</td>
</tr>
<tr>
<td>Pebble gr. impurities</td>
<td>0.639%</td>
<td>0.749%</td>
<td>1.126%</td>
<td>17.2%</td>
<td>50.3%</td>
</tr>
<tr>
<td>Pebble gr. temperature</td>
<td>0.694%</td>
<td>0.753%</td>
<td>0.972%</td>
<td>8.4%</td>
<td>29.1%</td>
</tr>
<tr>
<td>Structural gr. density</td>
<td>0.873%</td>
<td>0.952%</td>
<td>0.820%</td>
<td>9.1%</td>
<td>-13.9%</td>
</tr>
<tr>
<td>Structural gr. impurities</td>
<td>0.921%</td>
<td>1.109%</td>
<td>0.990%</td>
<td>20.3%</td>
<td>-10.7%</td>
</tr>
<tr>
<td>Structural gr. temperature</td>
<td>0.998%</td>
<td>1.135%</td>
<td>0.920%</td>
<td>13.7%</td>
<td>-18.9%</td>
</tr>
</tbody>
</table>
Results and Key Nuclear Data
(Subset of results from HTR-10 benchmark)
Uncertainty Analysis Results

Top Nuclear Data Contributors to Multiplication Factor Uncertainty
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

Evaluation Model and Suite of Codes

Code strategy for source term

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

Coming in 2022

Molten-salt fueled reactor
Sodium-cooled fast reactor
Molten Salt Reactor Experiment (MSRE)

Contract No. W-7405-eng-26
Reactor Division

MSR DESIGN AND OPERATIONS REPORT
PART I
DESCRIPTION OF REACTOR DESIGN
R. C. Robertson

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

JANUARY 1965

OASIS Slide 77

MSRE

Time = -10 sec
-0.17 min
dt = 50 ms

Spray flowrate 64.9 gpm
He purge 6.00×10^3 liters/day

Pump pressure 1.4 bar
Pump bowl 1231 F
Pump head 6.2 ft
Loop flowrate 1269 gpm

Fission power 8789.98 kW
Loop heating 777.34 kW
Graphite heating 943.45 kW
Total power 10000.00 kW
Fuel total reactivity 0.0 pcm
Control reactivity 0.0 pcm

Secondary flowrate 850.0 gpm
Inlet Temp 1025.0 F
Outlet temp 1100.5 F
Heat removal 15.5 MW

Core inlet temp 1174.4 F
Core outlet temp 1228.4 F
Core dp 5.2 psi
Core inlet P 19.3 psig

Drain flowrate 0.00 lb/min
Drain tank mass 6.0 lbm
Advanced Burner Test Reactor (ABTR)
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
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)
• 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.

SCALE analysis approach for SFR
MELCOR Modeling Scope

- Thermal hydraulics
- Reactivity Effects
- Fuel thermal-mechanical response
- Fission product release and transport
- Core degradation
- Ex-vessel damage progression
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
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
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 uncover through 24 hours

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

* With modifications by Ontario Tech.
Some accidents may involve reactivity feedbacks

For non-LWRs, MELCOR uses a point kinetics model

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
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
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

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

https://www.osti.gov/pages/servlets/purl/1437906
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
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
Directed Energy Deposition

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

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

https://www.osti.gov/pages/servlets/purl/1437906
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

• **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
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.

Expected to be developed later after DOE-EPRI demo project.
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

<table>
<thead>
<tr>
<th>Subtask</th>
<th>Actions/Deliverables</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>1A AMT processes under consideration</td>
<td>Additive Manufacturing (AM) – Laser Powder Bed Fusion</td>
<td>Complete - ML20351A292</td>
</tr>
<tr>
<td></td>
<td>AM – Directed Energy Deposition (DED)</td>
<td>Complete - ML20233A693</td>
</tr>
<tr>
<td></td>
<td>Cold Spray</td>
<td>Complete - ML21263A105</td>
</tr>
<tr>
<td></td>
<td>Powder Metallurgy (PM) – Hot Isostatic Pressing (HIP)</td>
<td>Draft report under NRC review</td>
</tr>
<tr>
<td></td>
<td>Electron Beam (EB) welding</td>
<td>Draft report under NRC review</td>
</tr>
<tr>
<td>1B Inspection and NDE</td>
<td>PNNL NDE gap analysis</td>
<td>Complete - ML20349A012</td>
</tr>
<tr>
<td>1C Modeling and Simulation of Microstructure</td>
<td>ANL M&amp;S gap analysis to predict microstructure</td>
<td>Complete - ML20269A301</td>
</tr>
<tr>
<td></td>
<td>ANL M&amp;S gap analysis to predict material performance</td>
<td>Complete - ML20350B550</td>
</tr>
</tbody>
</table>
# Status of Deliverables – Tasks 2 and 3

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