Public Meeting

- Telephone Bridge
  (888) 793-9929
  Passcode: 1770692

- Opportunities for public comments and questions at designated times

- Meeting on Regulatory Basis for Possible Changes to Physical Security Requirements at 2:30
Outline

- Introductions
- Modeling & Simulation (NRC)
- Interface Requirements for Staged Licensing (NIA)
- Developer Priorities & HALEU (NIC)

- Policy Issues, Industry Needs Assessment
  - TRISO topical report
  - Future Meetings

- Regulatory Basis Development for Possible Changes to Physical Security Requirements
Modeling & Simulation

- **DBE Confirmatory Analysis Code Suite for Non-LWRs** (S. Bajorek)
- **MELCOR non-LWR ACTIVITIES** (H. Esmaili)
- **Consequence Analysis (MACCS) Code Development Plan for Non-LWRs** (J. Barr)
Break

Meeting/Webinar will begin shortly

Telephone Bridge
(888) 793-9929
Passcode: 1770692
• Nuclear Innovation Alliance
  – Ashley Finan
  – Establishing Interface Requirements in Support of Staged Licensing
• Nuclear Industry Council
  – David Blee, NIC
    • Developer Priorities
  – Stephen Crowne, URENCO
    • Next Generation Nuclear Fuels
Lunch
Meeting/Webinar will begin at 1:00pm

Telephone Bridge
(888) 793-9929
Passcode: 1770692
Implementation Action Plans

<table>
<thead>
<tr>
<th>Strategy 1</th>
<th>Strategy 2</th>
<th>Strategy 3</th>
<th>Strategy 4</th>
<th>Strategy 5</th>
<th>Strategy 6</th>
</tr>
</thead>
</table>

- ONRL Molten Salt Reactor Training
- Identification & Assessment of Available Codes
- Regulatory Roadmap
- ASME BPVC Section III Division 5
- Siting near densely populated areas

- Knowledge Management
- Prototype Guidance
- ANS Standards 20.1, 20.2 30.2, 54.1
- Insurance and Liability

- Competency Modeling
- Non-LWR Design Criteria
- Non-LWR PRA Standard
- Consequence Based Security (SECY-18-0076)

- Updated HTGR and Fast Reactor Training
- Environmental Reviews
- Licensing Modernization Project
- EP for SMRs and ONTs (SECY-18-0103)

- Functional Containment (SECY-18-0096)

- Potential First Movers
- Micro-Reactors

✓ - Completed
1. Staff Training
2. Computer Code Assessments
3. Interactions with Licensing Modernization Project (DG 1353)
   Environmental Review Working Group
   Update Roadmap
4. ASME Div 5, ANS Design Standards, non-LWR PRA Standard
5. Policy Issues
   Siting, PAA, Security, EP, Functional Containment
6. Communications
7. “Micro-Reactors”
## Policy Table

<table>
<thead>
<tr>
<th></th>
<th>Ongoing Activities</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Prototype Guidance</td>
</tr>
<tr>
<td></td>
<td>Staged Licensing</td>
</tr>
<tr>
<td></td>
<td>Roadmap</td>
</tr>
<tr>
<td></td>
<td><em>(plan to update)</em></td>
</tr>
<tr>
<td>2a</td>
<td>Source Term</td>
</tr>
<tr>
<td></td>
<td>Prepare MST Guidance</td>
</tr>
<tr>
<td></td>
<td>Dose Calcs</td>
</tr>
<tr>
<td></td>
<td>Sitinng</td>
</tr>
<tr>
<td></td>
<td>Prepare Siting Guidance</td>
</tr>
<tr>
<td>2b</td>
<td>SSC Design Issues</td>
</tr>
<tr>
<td></td>
<td>NEI 18-04, DG-1353</td>
</tr>
<tr>
<td>3</td>
<td>Offsite EP</td>
</tr>
<tr>
<td></td>
<td>SECY-18-103</td>
</tr>
<tr>
<td>4</td>
<td>Insurance/Liability</td>
</tr>
<tr>
<td></td>
<td>Future <em>(2021)</em> Report to Congress</td>
</tr>
<tr>
<td></td>
<td><em>(no change acceptable)</em></td>
</tr>
<tr>
<td>5</td>
<td>PRA in licensing</td>
</tr>
<tr>
<td></td>
<td>NEI 18-04, DG-1353</td>
</tr>
<tr>
<td>6</td>
<td>Defense in Depth</td>
</tr>
<tr>
<td></td>
<td>NEI 18-04, DG-1353</td>
</tr>
<tr>
<td>7</td>
<td>Physical Security</td>
</tr>
<tr>
<td></td>
<td>SECY-18-0076</td>
</tr>
<tr>
<td></td>
<td><em>(limited to sabotage)</em></td>
</tr>
</tbody>
</table>
## Policy Table

### Ongoing Activities

<table>
<thead>
<tr>
<th></th>
<th>Activity</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>8</td>
<td>LBEs</td>
<td>NEI 18-04, DG-1353</td>
</tr>
<tr>
<td>9a</td>
<td>Fuel Qualification</td>
<td>technology specific</td>
</tr>
<tr>
<td>9b</td>
<td>Materials Qualification</td>
<td>technology specific</td>
</tr>
<tr>
<td>10a</td>
<td>MC&amp;A Cat II facilities</td>
<td>ML18267A184</td>
</tr>
<tr>
<td>10b</td>
<td>Security Cat II facilities</td>
<td>ML18267A184</td>
</tr>
<tr>
<td>10c</td>
<td>Collaboration</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• criticality benchmark</td>
<td></td>
</tr>
<tr>
<td></td>
<td>• HALEU shipping</td>
<td></td>
</tr>
<tr>
<td>11</td>
<td>Functional Containment Performance Criteria</td>
<td>SECY-18-0096 &amp; SRM</td>
</tr>
<tr>
<td>?</td>
<td>Advanced Manufacturing</td>
<td></td>
</tr>
</tbody>
</table>
Policy Table

<table>
<thead>
<tr>
<th></th>
<th>Open – Not Working</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Annual Fees</td>
</tr>
<tr>
<td>2</td>
<td>Manufacturing License</td>
</tr>
<tr>
<td>3</td>
<td>Process Heat</td>
</tr>
<tr>
<td>4</td>
<td>Waste Issues</td>
</tr>
</tbody>
</table>
| 5 | Operator Staffing*  
<p>|    | Remote/Autonomous |</p>
<table>
<thead>
<tr>
<th></th>
<th>No Plans (Resolved or Need Feedback)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Multi-module License</td>
</tr>
<tr>
<td>2</td>
<td>Operator Staffing*</td>
</tr>
<tr>
<td>3</td>
<td>Operational Programs</td>
</tr>
<tr>
<td>4</td>
<td>Module Installation</td>
</tr>
<tr>
<td>5</td>
<td>Decommissioning Funding</td>
</tr>
<tr>
<td>6</td>
<td>Aircraft Impact Assessments</td>
</tr>
</tbody>
</table>
NEI / ARRTF Updates
TRISO Topical Update
### 2019 Tentative Schedule; Periodic Stakeholder Meetings

<table>
<thead>
<tr>
<th>Date</th>
<th>Topic</th>
</tr>
</thead>
<tbody>
<tr>
<td>February 7</td>
<td>Civil/Structural Design/Licensing Issues (e.g., seismic isolation)</td>
</tr>
<tr>
<td>March 28</td>
<td></td>
</tr>
<tr>
<td>May 9</td>
<td></td>
</tr>
<tr>
<td>June 27</td>
<td></td>
</tr>
<tr>
<td>August 15</td>
<td></td>
</tr>
<tr>
<td>October 10</td>
<td></td>
</tr>
<tr>
<td>December 11</td>
<td></td>
</tr>
</tbody>
</table>
Break

Meeting/Webinar on *Regulatory Basis for Possible Rulemaking on Physical Security* will begin shortly

Telephone Bridge
(888) 793-9929
Passcode: 1770692
IAP Strategy 2: DBE Confirmatory Analysis Code Suite for Non-LWRs

Stephen M. Bajorek, Ph.D.
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission
Ph.: (301) 415-2345 / Stephen.Bajorek@nrc.gov

Advanced Reactor Stakeholder Meeting
December 13, 2018
“Strategy 2” Codes for Design Basis Events

- Numerous options available for thermal-hydraulics, neutronics, and fuel performance analysis for non-LWRs.
- Evaluation of codes for NRC use began with gaining a better understanding of the technologies. Existing PIRTs were augmented by new PIRTs developed for molten-salt reactors.
- “Hands-on” training and experience in DOE codes by NRC staff.
“Strategy 2” Codes for Design Basis Events

• Codes considered:
  – NRC legacy codes (TRACE, PARCS, FRAPCON, FAST)
  – DOE NEAMS codes (MAMMOTH, PRONGHORN, RELAP7)
  – ANL codes (SAS4A/SASSY, SAM, PROTEUS, MC2, Nek5000)
  – DOE CASL codes (MPACT, CTF, BISON, MAMBA)
  – Commercial codes (FLUENT, COMSOL)

• Recommended approach is to use a system of coupled codes, “Comprehensive Reactor Analysis Bundle” (CRAB). This includes codes from the NRC and DOE.
Code Selection Considerations

- **Physics.** Code suite must now or with development capture the correct physics to simulate non-LWRs. Selection of codes based on results of PIRTs. Code coupling necessary for “multi-physics”.

- **Flexibility.** Multiple reactor design concepts require flexibility within code suite. A goal has been to limit the number of new codes and need for staff training.

- **Code V&V.** Code assessment is critical, especially assessment relative to non-LWRs.

- **Computation Requirements.** Must be able to run simulations on HPC platforms available to NRC.

- **Cost avoidance.** An objective is to minimize cost to the NRC by leveraging DOE tools and influencing development plans.

Codes selected for CRAB satisfy these criteria.
DBE Analysis Codes

• Code Suite Report (draft) describes analysis approach for each of 10 distinct design types.
  – Gaps
  – Assessment
  – Tasks

• Reference plant models being developed.
Comprehensive Reactor Analysis Bundle (CRAB for LWRs)
Comprehensive Reactor Analysis Bundle (CRAB for LWRs w/ATF)

- SCALE: Cross-sections
- PARCS: Neutronics
- TRACE: System T/H
- FAST: Fuel Performance
- SERPENT: Cross-sections
- MAMMOTH: Neutronics
- PRONGHORN: Core T/H
- MOOSE
- BISON: Fuel Performance
- SAM: System and Core T/H
- MAMMOTH: Neutronics
- SERPENT: Cross-sections
- FLUENT: CFD
- Nek5000: CFD
- MELCOR: Containment / FP

NRC Code
Int’l Code
Commercial
DOE Code
Comprehensive Reactor Analysis Bundle (CRAB for GCRs)

- SCALE: Cross-sections
- PARCS: Neutronics
- TRACE: System T/H
- FAST: Fuel Performance
- SERPENT: Cross-sections
- MAMMOTH: Neutronics
- PRONGHORN: Core T/H
- MOOSE
- BISON: Fuel Performance
- SAM: System and Core T/H
- MAMMOTH: Neutronics
- SERPENT: Cross-sections
- FLUENT: CFD
- Nek5000: CFD
- MELCOR: Containment / FP

Codes:
- NRC Code
- Int'l Code
- Commercial
- DOE Code
Comprehensive Reactor Analysis Bundle (CRAB for Heat Pipe Reactors)

- SCALE Cross-sections
- PARCS Neutronics
- TRACE System T/H
- FAST Fuel Performance
- SERPENT Cross-sections
- MAMMOTH Neutronics
- PRONGHORN Core T/H
- MOOSE
- BISON Fuel Performance
- SAM System and Core T/H
- MAMMOTH Neutronics
- SERPENT Cross-sections
- FLUENT CFD
- Nek5000 CFD
- MELCOR Containment / FP
- NRC Code
- Int’l Code
- Commercial
- DOE Code

Slide 10
Code Readiness

- Using PCMM (Predictive Capability Maturity Model) to characterize code readiness.
  - Geometric Fidelity
  - Physics and Model Fidelity
  - Code Verification
  - Solution Verification
  - Code Validation
  - Uncertainty Quantification

- Rating scale “0” to “3”
  “D” “A”
Summary & Conclusions

- “Code Suite Report” recommends the codes in CRAB as the approach for non-LWR analysis.

- Using the combination of NRC and DOE codes will provide a **technically superior product** than can be attained with further development of the NRC’s legacy LWR codes only.

- Using the DOE codes provides a **significant benefit in resources & schedule** to the NRC. DOE has been cooperative in revising their plans to fit our needs and schedule.
MELCOR non-LWR ACTIVITIES

Hossein Esmaili
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

December 13, 2018
MELCOR Overview

MELCOR developed at Sandia National Laboratories for the U.S. NRC

- State-of-the-art tool for severe accident progression and source term analysis. Ongoing development of new capabilities
- Replace collection of simple, special purpose codes, i.e., Source Term Code Package (STCP)
- Eliminate tedious hand-coupling between modules
- Capture feedback effects (i.e., coupling of temperatures, release rates, and decay heating)
MELCOR Code Development

- Fully Integrated, engineering-level code
  - Thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings;
  - Core heat-up, degradation, and relocation;
  - Core-concrete attack;
  - Hydrogen production, transport, and combustion;
  - Fission product release and transport behavior

- Traditional Application
  - User constructs models from basic constructs
    - Control volumes, flow paths, heat structures,
  - Multiple ‘CORE’ designs
    - PWR, BWR, HTGR (Pebble Bed & PMR), PWR-SFP, BWR-SFP, SMR, Sodium (Containment)
  - Adaptability to new reactor designs

- Validated physical models
  - ISPs, benchmarks, experiments, accidents

- Uncertainty Analysis
  - Relatively fast-running
  - Characterized numerical variance

- User Convenience
  - Windows/Linux versions
  - Utilities for constructing input decks (GUI)
  - Capabilities for post-processing, visualization
  - Extensive documentation

- Non-LWR Reactors
  - HTGR/SFR/MSR

Code Development & Regulatory Applications

- State-of-the-Art Reactor Consequence Analysis (SOARCA)
- Site Level 3 Probabilistic Risk Assessment
- Spent Fuel Pool Study (NUREG-2161) Tier 3 Expedited Fuel Transfer (COMSECY-13-0030)
- Independent Spent Fuel Storage Installation Rulemaking
- Fukushima accident forensic analysis and reconstruction (DOE/NRC; BSFNEA)
- Fukushima Near-Term Task Force Recommendations 5.1, 5.2, 5.6
  - Filtered containment venting (BWR Mark I & II)
- System success criteria (SPAR)
- Severe accident induced steam generator tube rupture
- NUREG-1465 source term validation
- Revised source term for high burnup and mixed oxide fuel
- Severe accident management
- New and Advanced reactor design licensing (IPWR)
- Canadian MOX/LTA licensing
- Risk-Informed 10 CFR 50.48
- Risk-Informed regulation (NB/MMSS)

International Collaboration (CSARP/MCAP/EMUG/AMUG)

Integrated models required for self-consistent analysis
Non-LWR Beyond Design Basis Events

- Development of evaluation models (example HTGR)
  - ACRS Future Plant Designs Subcommittee, April 5, 2011
SCALE Code & Application to MELCOR/MACCS

- Oak Ridge Isotope Generation code (ORIGEN)
- Irradiation and decay simulation code
- Fuel depletion and used fuel characterization
- Source terms for accident analyses (operating reactors, spent fuel handling, storage, etc.)
- Structural material activation (in-core, ex-core)
- Material feed and removal for fuel cycle and liquid fuel
- ORIGEN data enable comprehensive isotopic characterization of fuel over a large time scale, including repository analysis

ENDF/B
Physics data
Thermal scattering law, resonance data, energy distributions, fission yields, decay constants, etc.

AMPX
Validated cross section libraries; depletion and decay data

TRITON / Polaris
Transport and depletion in 1D, 2D, and 3D for LWR, ATF, and nonLWR

ORIGEN / ORIGAMI
Depletion, activation and decay
Reactor-specific radioactive source term characterization
High Temperature Gas Cooled Reactors

Existing Modeling Capabilities

- Helium Properties
- Accelerated steady-state initialization
- Two-sided reflector (RF) component
- Modified clad (CL) component (PMR/PBR)
- Core conduction
- Point kinetics
- Fission product diffusion, transport, and release
- TRISO fuel failure

Graphite dust transport
  - Turbulent deposition, Resuspension

Basic balance-of-plant models (Turbomachinery, Heat exchangers)

Momentum exchange between adjacent flow paths (lock-exchange air ingress)

Graphite oxidation

\[
\frac{\partial C}{\partial t} = \frac{1}{r^m} \frac{\partial}{\partial r} \left( r^m D \frac{\partial C}{\partial r} \right) - \lambda C + S
\]

Modeling Gaps

- Current modeling uses UO2 material properties, needs to be extended to UCO
Molten Salt Reactors

- Properties for LiF-BeF2 have been added
  - Equation of State
    - Current capability
  - Thermal-mechanical properties
    - Current capability
  - EOS for other molten salt fluids would need to be developed
    - Minor modeling gap
- Fission product modeling
  - Fission product interaction with coolant, speciation, vaporization, and chemistry
    - Moderate modeling gap
- Two reactor types envisioned
  - Fixed fuel geometry
    - TRISO fuel models
      - Current capability
  - Liquid fuel geometry
    - MELCOR CVH/RN package can model flow of coolant and advection of internal heat source with minimal changes.
      - Current capability
    - COR package representation no longer applicable but structures can be represented by HS package
    - Calculation of neutronics kinetics for flowing fuel
      - Significant modeling gap.
Sodium Fast Reactors

Existing Modeling Capabilities

- Sodium Properties
  - Sodium Equation of State
  - Sodium Thermo-mechanical properties
- Containment Modeling
  - Sodium pool fire model
  - Sodium spray fire model
  - Atmospheric chemistry model
  - Sodium-concrete interaction model

Modeling Gaps

- SFR Core modeling
  - Fuel thermal-mechanical properties
  - Fuel fission product release and transport
    - FP speciation & chemistry
    - Bubble transport through a sodium pool
  - Core degradation models
    - SASS4A surrogate model
    - Heat pipe specific models
- Containment Modeling
  - Capability for having more than one working fluid
  - Vaporization rates of RNs from sodium pool surface
  - Radionuclide entrainment near pool surface during fires
    - Transport of FP in sodium drops
  - Hot gas layer formation during sodium fires
  - Oxygen entrainment into a pool fire
  - Sodium water reactions
Design Basis Source Term Development Process
(example: MOX & High Burnup Fuel)

Experimental Basis
- Oxidation/Gas Generation
- Melt Progression
- Fission Product Release
- Fission Product Transport

PIRT process

Accident Analysis
- Scenario # 1
- Scenario # 2
- Scenario # n-1
- Scenario # n

Synthesize timings and release fractions

Design Basis Source Term

- Similar RFs to NUREG-1465 but prolonged release
- Differences not from change of fuel but from code advances

Consequence Analysis (MACCS)
Code Development Plan for Non-LWRs

Jonathan Barr
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

December 13, 2018
MACCS Overview

• MACCS is the only code used in U.S. for probabilistic offsite consequence analysis
• Treats all technical elements of Level 3 PRA standard: radionuclide release, atmospheric transport, meteorology, protective actions, site data, dosimetry, health effects, economic factors, uncertainty
MACCS Overview

• Highly flexible code enabling applicability to different types of sources and accidents
• Variety of associated risk measures
  – Dose
  – Radiological health effects and fatality risk
  – Economic impact
  – Land contamination
  – Population affected by protective actions
• Developed by NRC over 3+ decades
• MACCS recently has been used in major studies including State-of-the-Art Reactor Consequence Analyses (SOARCA), Level 3 PRA project, and various Fukushima-related applications
• Part of Cooperative Severe Accident Research Program (CSARP) with 28 member countries
MACCS Applications

- Regulatory cost-benefit analysis
- Environmental report analyses of Severe Accident Mitigation Alternatives (SAMA) and Design Alternatives (SAMDA)
- Level 3 PRA
- Research studies of accident consequences
- Support for emergency preparedness
- Dose-distance evaluations for emergency planning
MACCS for Non-LWRs

• Code development plans for design-specific issues
  – Radionuclide screening
  – Radionuclide size
  – Radionuclide chemical form
  – Radionuclide shape factor
  – Tritium

• Code development plans for site-related issues
  – Near-field atmospheric transport
  – Decontamination modeling
Near-Field Atmospheric Transport

- MACCS currently has a simple model for building wake effects; user guide cautions against use closer than 500m.
- Non-LWRs (and SMRs) desire smaller EPZ and site boundary than large LWRs; therefore desire better modeling of near-field phenomena.

Near-Field Atmospheric Transport

- Various options for addressing near-field ATD
  - Modifications to Gaussian plume segment ATD model
  - CFD modeling of 3-d wind field with Lagrangian particle tracking ATD model
  - Empirical models of 3-d wind fields with Lagrangian particle tracking ATD model

- Considerations for evaluating options
  - Extent of practical acceptance in the user community
  - Simplicity of use
  - Computational efficiency
  - Cost and time efficiency
  - Accuracy
  - Feasibility for probabilistic application

Example QUIC-URB simulation of wind vectors
Example QUIC-PLUME simulation of urban transport and dispersion
Background Documents

- 10 CFR Part 52, Subpart E allows an applicant to seek standard design approval for either an entire plant or “major portions” thereof
- NRC staff provided feedback on this report on July 20, 2017 (ML17201Q109)

Table of Contents:

Executive Summary
Introduction
Purpose and Scope
Standard Design Approval
Methods to Develop Interface Requirements
Example Cases
  Core Design
  Reactor Vessel Auxiliary Cooling System Design
  Reactor Coolant System Piping Design
  Reactor Building Structural Design
Conclusions
Introduction

• Many companies are developing new designs with new safety approaches
• Some companies are using predominantly private funding, and thus confront different investment requirements from historic projects
• Companies will take a variety of licensing approaches appropriate to their business plan
Figure 1: Current Project Risk/Investment Profile Relative to Detailed Design & Licensing

Figure 2: Desirable Project Risk/Investment Profile Relative to Detailed Design & Licensing
Staged Licensing Review Approach

• Some companies may opt for a staged review approach using any of:
  – Licensing project plan or regulatory engagement plan
  – Preliminary design reviews
  – Topical and/or technical reports
  – Standard design approval
  – Construction permit or design certification
Purpose and Scope

• Provide guidance to vendors using the SDA on the establishment of interface requirements between portions of a design in the SDA with those that will be submitted at a later date

• Any reactor type
Standard Design Approval

- 10 CFR Part 52 Subpart E
  - Documents staff findings, involves ACRS reviews, provides reference for subsequent applications
  - Incremental progress towards licensing or certification as part of staged licensing
- Potential value:
  - Licensing risk reduction (via approval of limited portion of design)
  - Reduce initial development cost (defer portions to subsequent licensing steps)
  - Approval for portion as part of commercial strategy, e.g.:
    - Optional design features such as power uprate or non-electric application
    - Deployment outside US
- May result in greater overall cost/timeline compared with single successful application
Methods to Develop Interface Requirements

• Have approved QA program
• Clearly define scope of SDA
  – SSCs, engineering disciplines, technical bases for satisfying principal design criteria (PDC)
    • PDC could be derived from Reg Guide 1.232, for example, or the LMP guidelines.
• Set boundary conditions with functional and operational characteristics of SSCs that are not within scope
  – These will have to be satisfied in subsequent submittals, if full design approval is sought
  – Margins are required; size of margins may impact economics
Process for Developing Interface Requirements in Support of an SDA

1. Perform Conceptual Design
2. Define SDA Scope
3. Develop Design Requirements
4. Complete Major Portion of Design
5. Develop Interface Requirements
6. Applicable ARDC and Regulations
Example Cases

• Core Design
• Reactor Vessel Auxiliary Cooling System Design
• Reactor Coolant System Piping Design
• Reactor Building Structural Design

• Tables delineate interface requirements of the SDA example and are organized by ARDC
Example: RVAC System Interface

Requirements

- Quality standards and records
- Design basis for protection against natural phenomena
- Fire protection
- Environmental and dynamic effects design bases
- Instrumentation and control
- Containment design
- Protection system functions
- Residual heat removal
- Emergency core cooling
- Containment heat removal
- Inspection of containment heat removal system
- Testing of containment heat removal system
- Containment design basis
<table>
<thead>
<tr>
<th>ARDC</th>
<th>Title</th>
<th>Sample Interface Requirements for RVAC System</th>
</tr>
</thead>
<tbody>
<tr>
<td>2</td>
<td>Design basis for protection against natural phenomena</td>
<td>Interface Requirement</td>
</tr>
<tr>
<td></td>
<td>The ability of the SSCs of the RVAC to withstand the design basis</td>
<td>The comparison of the FSAR design assumptions</td>
</tr>
<tr>
<td></td>
<td>natural phenomena will be addressed in the FSAR. The comparison of</td>
<td>to those relating to an actual site will be</td>
</tr>
<tr>
<td></td>
<td>the FSAR design assumptions to those relating to an actual site will</td>
<td>addressed in a future submittal. Adequate</td>
</tr>
<tr>
<td></td>
<td>be addressed in a future submittal. Adequate margin should be</td>
<td>margin should be included in the assumed</td>
</tr>
<tr>
<td></td>
<td>included in the assumed values for the natural phenomena to provide</td>
<td>values for the natural phenomena to provide</td>
</tr>
<tr>
<td></td>
<td>flexibility in siting the design.</td>
<td>flexibility in siting the design.</td>
</tr>
<tr>
<td></td>
<td>The FSAR will specify seismic, hurricane, and tornado design</td>
<td></td>
</tr>
<tr>
<td></td>
<td>parameters (e.g., earthquake design response spectra, soil</td>
<td></td>
</tr>
<tr>
<td></td>
<td>conditions, tornado and hurricane wind speeds, etc.). These</td>
<td></td>
</tr>
<tr>
<td></td>
<td>parameters will be compared to those evaluated for a future site.</td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>Fire protection</td>
<td>Interface Requirement</td>
</tr>
<tr>
<td></td>
<td>The RVAC is required to have a fire protection program. The fire</td>
<td>The RVAC is required to have a fire</td>
</tr>
<tr>
<td></td>
<td>protection program will be addressed in a future submittal.</td>
<td>protection program will be addressed in a</td>
</tr>
<tr>
<td></td>
<td>The FSAR will include a commitment that the materials used in the</td>
<td>future submittal.</td>
</tr>
<tr>
<td></td>
<td>RVAC structure will use noncombustible and fire-resistant materials</td>
<td>The FSAR will include a commitment that the</td>
</tr>
<tr>
<td></td>
<td>wherever practical, particularly in locations with SSCs important to</td>
<td>materials used in the RVAC structure will</td>
</tr>
<tr>
<td></td>
<td>safety.</td>
<td>use noncombustible and fire-resistant</td>
</tr>
<tr>
<td></td>
<td></td>
<td>materials wherever practical, particularly</td>
</tr>
<tr>
<td></td>
<td></td>
<td>in locations with SSCs important to safety.</td>
</tr>
</tbody>
</table>
Next Steps

• Q&A today
• Feedback factored into revised report
• NRC Feedback

Thank you!
Thank you

Feedback & Questions

Please feel welcome to send additional input at any time to Ashley Finan (ashley@nuclearinnovationalliance.org).
Priorities for Advanced Reactor Developers:

USNIC Survey of Developer Priorities

December 13, 2018

David Blee
President & CEO
U.S. Nuclear Industry Council

Hon. Jeffrey S. Merrifield
Former Commissioner, USNRC;
Chairman, USNIC Advanced Reactors Task Force;
Partner, Pillsbury Winthrop Shaw Pittman
USNIC AR Developers Survey

- USNIC conducted a third in a series survey of 16 leading U.S. Advanced Reactor technology developers with regard to DOE Initiatives

- 15 Developers responded, one respondent per company

- This was a blind survey so individual results were not identified
Survey Goals

- Intended to provide stakeholder feedback on NRC preparations for Advanced Reactor Licensing
- Feedback is intended to give constructive input to the Commission and Staff
- Survey provides a snapshot of the current policy priorities of the Advanced Reactor Community
- Assessment goes beyond the efforts of the Office of New Reactors to include the preparations of other NRC offices
- Provides feedback on the perceived technical readiness of the NRC staff
Q1: Pace of the NRC’s Advanced Reactor Licensing Transformation: Rate the pace of the NRC’s Preparation Activities for Advanced Reactor licensing?
Q2: NRC Support for Advanced Reactor Licensing Transformation: Please rank the NRC Offices' prioritization of Advanced Reactor transformation?

<table>
<thead>
<tr>
<th>Office</th>
<th>Clearly a high priority</th>
<th>Appears to be a priority</th>
<th>Not enough of a priority</th>
</tr>
</thead>
<tbody>
<tr>
<td>NRC Chairman &amp; Commissioners</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Office of New Reactors</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Office of Nuclear Material Safety and Safeguards</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Office of Nuclear Security and Incident Response</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Legend:
- Green: Clearly a high priority
- Blue: Appears to be a priority
- Orange: Not enough of a priority
Q3: Planning Timeframe for Licensing Application Submittals: What should the NRC and DOE’s Planning Timeframe be for new Advanced Reactor License Applications?
Q4: Focus for NRC Advanced Reactors Licensing Transformation in 2019: What should the NRC’s key Licensing Transformation Focus be in? (ranked)
Q5: Early Resolution of NRC Policy Issues (e.g. emergency preparedness, consequence-based physical security): How do you think the NRC is doing with respect to resolving Key Policy issues early?
Q6: Enhanced Pre-Licensing Engagement: What actions would most improve the NRC’s pre-licensing engagement (rank in order of priority)?

- Cost-share for pre-licensing
- Fixed price and schedule certainty for pre-licensing
- Enhanced NRC Advanced Reactor Technology capability
- More robust stakeholder engagement
- Additional involvement by the Office of New Reactors
- Additional involvement by the Office of Nuclear Security & Incident Response
- Additional involvement by the Office of Nuclear Material, Safety & Safeguards
Q7: NRC Advanced Reactors Technical Capability: Please rate the NRC’s Advanced Reactor technology technical capability?
Q8: Confidence in NRC Advanced Reactors Licensing Schedule and Cost: What is your confidence that the NRC can transform its licensing process to provide greater schedule and cost certainty?
Q9: Should the NRC be doing more to seek non-fee based funding?
Q10: Value of NRC Advanced Reactor Stakeholder Meetings: Are the NRC’s Stakeholder Meetings (held every 6-8 weeks)?

- Useful
- Need more focus and...
- Be more frequent
- Be less frequent
- Other (please specify)
Q11: Do you believe the NRC Office of Research is putting sufficient time and resources towards Advanced Reactor development?
Q12: Versatile Advanced Test Reactor: How important is the deployment of a new U.S. Department of Energy advanced test reactor (Versatile Test Reactor) by 2026?
Summary Results

• Commission and staff of Office of New Reactors are perceived as making progress on Advanced Reactor policy decisions and licensing readiness

• Office of Nuclear Materials Safety and Safeguards and to a somewhat lesser extent the Office of Nuclear Security and Incident Response are not perceived as having the same level of engagement on Advanced Reactor issues

• Agency readiness for High Temperature Reactors is very good

• Higher level of questioning about NRC readiness to license Molten Salt, Fast and Liquid Metal Reactors

• There is a lack of understanding of what the Office of Research is doing to assist in preparing the NRC for Advanced Reactors

• There was an overwhelming view that the Commission needs to do more to assist in lifting the burden of Fee Based programs on Advanced Reactors
The United States Nuclear Industry Council (USNIC) is the leading U.S. business consortium advocate for nuclear energy and promotion of the American supply chain globally. Composed of over 80 companies USNIC represents the "Who’s Who" of the nuclear supply chain community, including key utility movers, technology developers, construction engineers, manufacturers and service providers. USNIC encompasses eight working groups and select task forces. For more information visit www.usnic.org
Meeting on Possible Regulatory Process Improvements for Advanced Reactors
December 13, 2018

Next Generation Nuclear Fuels

Stephen Cowne, Chief Nuclear Officer, UUSA
Today’s Front-End Nuclear Fuel Cycle

LWR Fuels
LEU-UO$_2$-ZircAlloy
Next Generation Fuel Pathways: Range of options

- Enrichment LEU\(^{+}\) Plus (5~10%)
- Higher Enrichment HA-LEU ~19.75%
- Existing UO\(_2\) Fuel Pellets • ~5.95% Enrichment
- ATF High Density Fuel Pellets • U-Silicide • U-Nitride • Chromium doped UO\(_2\) • FCM Ceramic Fuel
- ATF Cladding Systems • Chromium coating • Silicon-carbide cladding
- Gen-III Reactor Upgrades
- Test & Research Reactors
- DOE Programme Accident Tolerant Fuel
- Site Licensing Cat-II Facility
- Operational Criticality & Safety
- Intrinsically Safe Fuels
- National Regulator(s)
- Fuel Fabrication
- Storage & Transport • Cylinders • Overpacks • Class 7 Shipping • Insurance
- Govt., Licensing & Regulation
- Supply Chain
- Metal Cladding Systems • Lightbridge Zr-U Alloy • U-Molybdenum
- Liquid Fuels • Molten salts • Aqueous uranyl salt solutions
- RepU
- TRISO Fuel • UCO • UO\(_2\) • Uranium Nitride • Uranium Silicide
- Fabricated TRISO • Prismatic Block • Pebble Bed
- Metallic Fuel
- Lead Cooled Reactors
- Sodium Fast Reactors
- HTGR
- Molten Salt Reactors
- SMRs
- Micro-SMRs
- Fast Breeder Reactors
- HTGR
- Gen-III Reactor Uprates
- Reactor Types
- SMRs
- Storage & Transport
- Fuel Fabrication
- Supply Chain
- Government, Licensing & Regulation
- DOE Programme Accident Tolerant Fuel
- Site Licensing Cat-II Facility
- Operational Criticality & Safety
- Intrinsically Safe Fuels
- National Regulator(s)
The Future Nuclear Fuel Supply Chain

Existing Nuclear Fuel Supply Chain

- Mining: U₃O₈
- Conversion: 0.711% UF₆
- Enrichment: <5% LEU
- Fabrication: UO₂
- Back End: Spent Fuel

Completing the Future Nuclear Fuel Supply Chain

- Enrichment: 0.711% UF₆, <5% LEU
- Higher Enrichment: 5%-20% HA-LEU
- Deconversion: U-metal, U-oxide, U-salts
- Fabrication: Next Generation Fuels
  - TRISO (UCO), Uranium Nitride, Uranium Silicide, U-metal Alloys
  - UF₄ Salts etc...

LWR Reactors
- UO₂ / ZircAlloy Fuels
- Gen III+, ATF SMRs, GenIV
- Advanced Reactors
- Research & Test Reactors
HA-LEU and the HA-LEU Community

- **High Assay - Low Enriched Uranium (HA-LEU)** refers to enrichments above 5.0% U235 and below 20.0% U235.

- **A broad community of users may benefit from HA-LEU:**
  - Research & Test Reactors
  - Operators of existing LWRs seeking improvements in fuel reliability and economics through higher burnup and extended operating cycles
  - Accident Tolerant Fuels
  - Gen IV and other Advanced reactor designs
  - Advanced fuel designs
  - Producers of targets for medical isotope production

- **Fuel solutions are needed across the full span of HALEU enrichments**
  - some “clumping” may develop in the ranges of 6.0%-8.0% U235 and 13.0-16.0% U235 and at 19.75% U235.
A complete and sustainable HA-LEU fuel cycle includes three fundamental capabilities:

1. A Higher Enrichment Facility to produce HA-LEU enrichments:
   - the material will be in the form of uranium hexafluoride (UF6)
2. A conversion facility to (de)convert HA-LEU UF6 into metal, oxide and/or salts
3. One or more fabrication facilities that can manufacture the specific fuel types required by the various reactor and fuel designs

Packaging and transportation solutions are needed between each of these processing steps and to the ultimate user:

- Spent fuel packaging will also need to be considered at the back-end of the fuel cycle
Transport & Packaging Considerations

Existing UF6 Cylinders for Higher Assays (ANSI N14.1)

<table>
<thead>
<tr>
<th>Cylinder Model</th>
<th>Diameter (inches / mm)</th>
<th>Maximum Enrichment</th>
<th>Maximum UF6 (lbs / kgs)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1S</td>
<td>1.5 / 38.1</td>
<td>100.00%</td>
<td>1.0 / 0.5</td>
</tr>
<tr>
<td>2S</td>
<td>3.5 / 88.9</td>
<td>100.00%</td>
<td>4.9 / 2.2</td>
</tr>
<tr>
<td>5B</td>
<td>5.0 / 127</td>
<td>100.00%</td>
<td>54.9 / 24.9</td>
</tr>
<tr>
<td>8A</td>
<td>8.0 / 203.2</td>
<td>12.5%</td>
<td>255 / 115.7</td>
</tr>
<tr>
<td>30B</td>
<td>30 / 762</td>
<td>5%</td>
<td>5020 / 2277</td>
</tr>
</tbody>
</table>

- Are HA-LEU UF6 shipments limited to use of a small packaging?
- Are moderator exclusion requirements met through the cylinder or through an overpack?
- Criticality benchmarking data is needed for HA-LEU assays.
2-Box Model: Co-location of Enrichment & Deconversion

Problem:

- There is currently no available “transport package” for HA-LEU.

Possible Solution: “2-Box” Model: Co-location of Higher Enrichment and Deconversion Facilities.

- Reduces expense and time required to develop packaging and transport solutions
- Can be expanded to include fabrication facilities
- Satisfying the requirements of a number next generation fuel types for HA-LEU.
- Leverages existing site characterization data, site infrastructure, and regulator familiarity
HA-LEU Fuel Cycle: Licensing Approach

1a. Enrichments up to 5.5%
   • UUSA safety basis is analyzed at 6%, UUSA would need to demonstrate the reduction in the margin of safety to increase enrichment level limit.
     – Could be done quickly

1b. Enrichments above 5.5%
   • UUSA would need to reanalyze the design safety basis at higher enrichments
     – Analysis would require additional resources and will take more time.
   • CAT 2 – Changes to FNMCP and Security Plan
   • Level of effort required to achieve 19.75% limit vs. 7.0% limit is not that great.

2a. Utilizing existing transport packages for UF₆ above 5%
   • Criticality benchmarking data is needed for HA-LEU assays
   • For use with UO₂ fuel pellets

2b. UF₆ deconversion
   • For other fuel types
   • If existing transport packages are not approved at higher enrichments
1. **NRC resources and priorities**—due to the reductions in licensing staff at the NRC, the ability to review a license amendment in a timely manner is a concern. NRC should prioritize appropriately.

2. **Key rulemaking activities**
   - Part 50.68 change to support power industry
   - Part 171 Fees – new category for combined fuel cycle facility
   - Part 171 Fees – new category for moderate strategic SNM facility
   - Part 73 – highly diluted category

3. NRC must resist the temptation to revisit issues they want to change but are not required to raise enrichment limits. If analytical models are approved for licensees, there is no need to change.

4. Analytical codes are well validated up to 6%. Would need additional validation beyond 6%.
1. It is imperative that the enrichment, conversion and fabrication facilities - and the concordant packaging solutions - be developed on concurrent schedules.

2. The licensing framework needs to support development of a HA-LEU fuel cycle and regulator resources are needed.

3. Companies making investments in HA-LEU facilities need to be sufficiently assured of an economic return.

4. URENCO USA could submit a License Amendment Request (LAR) for 5.5% enrichment limit by April 30, 2019. A 6% LAR could be ready by June 30, 2019.

5. We all must “hold hands and jump together!”
URENCO: An Integrated Supplier

Thank You
SECY-18-0076
OPTIONS AND RECOMMENDATION FOR PHYSICAL SECURITY FOR ADVANCED REACTORS

December 13, 2018
NRC Advanced Reactor Policy Statement – Attributes:

• Highly reliable and less complex decay heat removal systems;
• Longer time constants to reaching safety system challenges;
• Simplified safety systems that reduce required operator actions;
• Designs that minimize the potential for severe accidents and their consequences; and
• Designs that incorporate the defense-in-depth philosophy by maintaining multiple barriers against radiation release.
Background

NRC Advanced Reactor Policy Statement

- Designs that include considerations for safety and security requirements together in the design process such that security issues (e.g., newly identified threats of terrorist attacks) can be effectively resolved through facility design and engineered security features, and formulation of mitigation measures, with reduced reliance on human actions.

- Challenge is to address policy issues related to how safety and security requirements for advanced reactors should reflect inherent design characteristics such as longer time constants before degradation of barriers and release of radioactive material given a loss of safety functions.
Background

  - The staff’s assessment determined that the current security regulatory framework is adequate to certify, approve, and license iPWRs …
  - The current regulations allow SMR designers and potential applicants to propose alternative methods or approaches to meet the performance-based and prescriptive security and MC&A requirements.
    - Alternate Measures (10 CFR 73.55(r))
    - License Conditions
    - Exemptions

- “The question at hand is whether some type of generic regulatory action would be preferable to the case-by-case approach described in SECY-11-0184.”
Identifies 4 Options:

1) No change / Status quo

2) Address possible requests for alternatives via guidance

3) Limited scope rulemaking to address what would otherwise be likely requests for alternatives

4) Broader based rulemaking to more fully reflect attributes of advanced reactors
Option 3 – Limited Scope Rulemaking

• Revise specific regulations and guidance related to physical security for SMRs and non-LWRs through rulemaking.
  o Example – NEI proposal for reductions in the number of armed responders (10 CFR 73.55(k)(5))

• NRC staff would interact with stakeholders to identify specific requirements within existing regulations that may play a diminished role in providing physical security for SMRs and non-LWRs while contributing significantly to capital or operating costs.

• NRC staff would develop guidance documents to support the implementation of the requirements defined through the rulemaking.
SRM Dated November 19, 2018

The Commission approved the staff’s recommended Option 3, to initiate a limited-scope revision of regulations and guidance related to physical security for advanced reactors and approved the enclosed rulemaking plan, subject to the enclosed edits.

- Complete regulatory basis — 12 months following Commission’s SRM

- Another potential area is the prescriptive requirements in 10 CFR 73.55 for onsite secondary alarm stations.
Rulemaking Process
Barrier Assessment (Bow Tie Diagram)

Note that top level event generally aligns with security concerns for radiological sabotage; a rulemaking, if pursued, would also need to address threats related to theft/diversion
Possible Performance (Consequence) Based Approach

NEI Proposed Logic for Applicability of Alternate Regulations (Armed Responders Not Required)

1. Uses a reactor technology that is not susceptible to significant core damage and spent fuel sabotage
   - Y: Alternate Regulation Applicable (Armed Responders Not Required)
   - N: Next Step

2. The facility design precludes the loss of safety functions and barriers to the release of fission products from the reactor core or spent fuel by threats up to and including the design basis threat
   - Y: Alternate Regulation Applicable (Armed Responders Not Required)
   - N: Next Step

3. Has engineered safety and security features that allow for implementation of mitigation strategies to prevent significant core damage and spent fuel sabotage if a target set is compromised, destroyed, or rendered nonfunctional (with local law enforcement assistance)
   - Y: Alternate Regulation Applicable (Armed Responders Not Required)
   - N: Alternate Regulation Not Applicable (Apply existing requirements or request exemption with design/site specific plan)
Security Design Considerations

Preliminary Draft Guidance (March 2017)

- Intrusion Detection Systems
- Intrusion Assessment Systems
- Security Communication Systems
- Security Delay Systems
- Security Response
- Control Measures for land/waterborne vehicle bombs
- Access Control Portals
- Cyber Security
Discussion

Potential Scope of Alternative Requirements

- 10 CFR 73.55(k) – armed responders
- 10 CFR 73.55(i) – secondary alarm stations
- ?
- ?
- ?
Stakeholder Presentation/Discussion
NEI
Discussion

Stakeholder Presentation/Discussion
USUCS
General Discussion

Public Questions/Feedback