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

Public Meeting on Possible Regulatory Process Improvements for Advanced Reactor Designs

December 13, 2018

Telephone Bridge
(888) 793-9929
Passcode: 1770692

Advanced Reactor Program

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

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*

Advanced Reactor Program

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

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

Implementation Action Plans

Strategy 1
Knowledge, Skills and Capability
- ✓ ONRL Molten Salt Reactor Training
- Knowledge Management
- ✓ Competency Modeling
- Updated HTGR and Fast Reactor Training

Strategy 2
Computer Codes & Review Tools
- Identification & Assessment of Available Codes

Strategy 3
Flexible Review Processes
- ✓ Regulatory Roadmap
- ✓ Prototype Guidance
- ✓ Non-LWR Design Criteria
- Environmental Reviews
- Licensing Modernization Project

Strategy 4
Consensus Codes and Standards
- ASME BPVC Section III Division 5
- ANS Standards 20.1, 20.2 30.2, 54.1
- Non-LWR PRA Standard

Strategy 5
Policy and Key Technical Issues
- Siting near densely populated areas
- Insurance and Liability
- Consequence Based Security (SECY-18-0076)
- EP for SMRs and ONTs (SECY-18-0103)
- Functional Containment (SECY-18-0096) ✓

Strategy 6
Communication
- ✓ NRC DOE Workshops
- Periodic Stakeholder Meetings
- NRC DOE GAIN MOU
- International Coordination

✓ - Completed

- Potential First Movers
- Micro-Reactors

Advanced Reactor Program

NRC Status

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

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

Policy Table

Ongoing Activities		
8	LBEs	NEI 18-04, DG-1353
9a	Fuel Qualification	technology specific
9b	Materials Qualification	technology specific
10a	MC&A Cat II facilities	ML18267A184
10b	Security Cat II facilities	ML18267A184
10c	Collaboration • criticality benchmark • HALEU shipping	
11	Functional Containment Performance Criteria	SECY-18-0096 & SRM
?	Advanced Manufacturing	

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

Policy Table

Open – Not Working		
1	Annual Fees	
2	Manufacturing License	
3	Process Heat	
4	Waste Issues	
5	Operator Staffing* Remote/Autonomous	

Advanced Reactor Program

Policy Table

No Plans (Resolved or Need Feedback)		
1	Multi-module License	
2	Operator Staffing*	
3	Operational Programs	
4	Module Installation	
5	Decommissioning Funding	
6	Aircraft Impact Assessments	

NEI / ARRTF Updates

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

TRISO Topical Update

Advanced Reactor Program

Future Meetings

2019 Tentative Schedule; Periodic Stakeholder Meetings	
February 7	Civil/Structural Design/Licensing Issues (e.g., seismic isolation)
March 28	
May 9	
June 27	
August 15	
October 10	
December 11	

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.

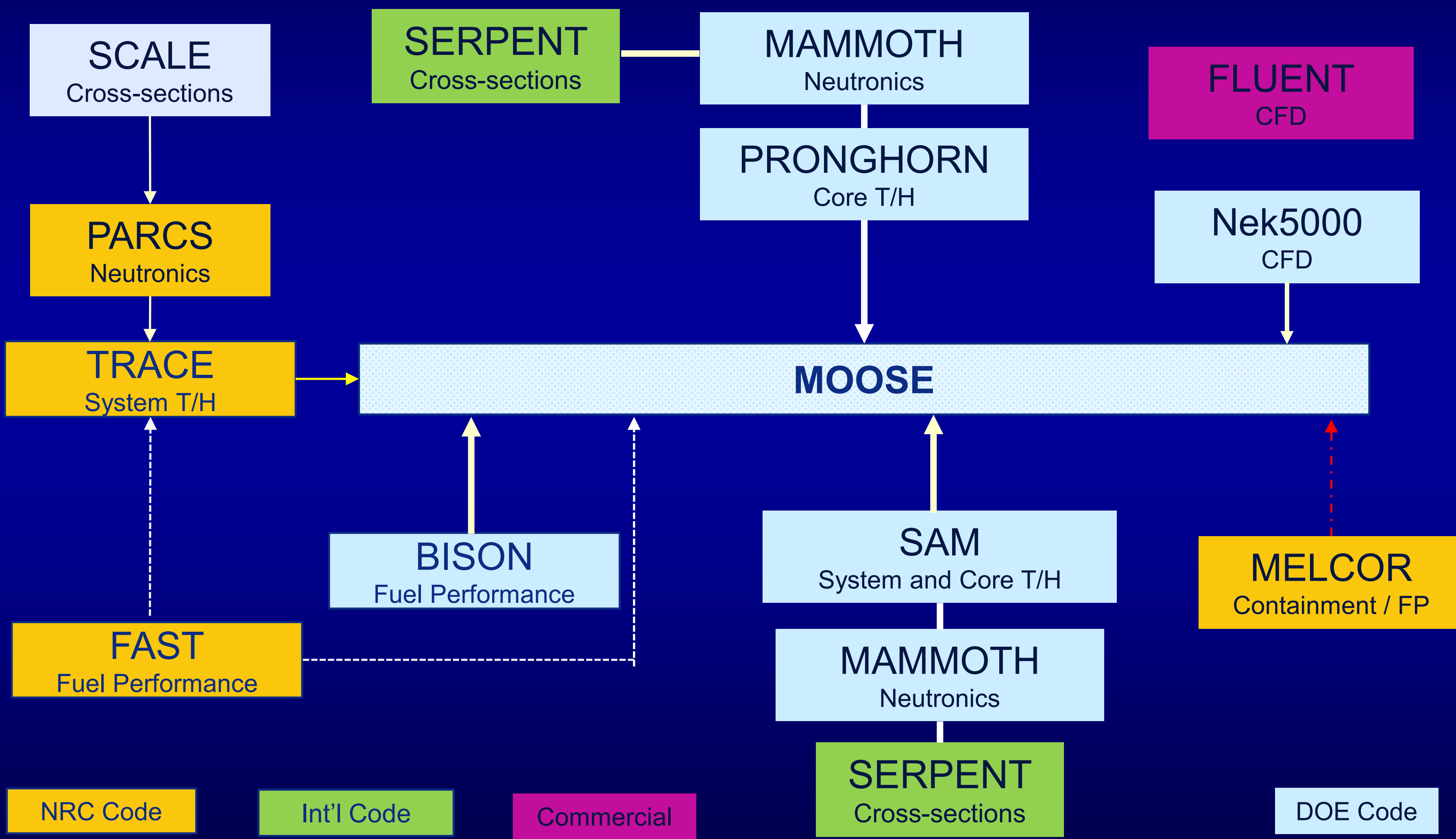
“Strategy 2” Codes for Design Basis Events

- 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.

Comprehensive Reactor Analysis Bundle (CRAB)

Current View; Oct.2018



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.

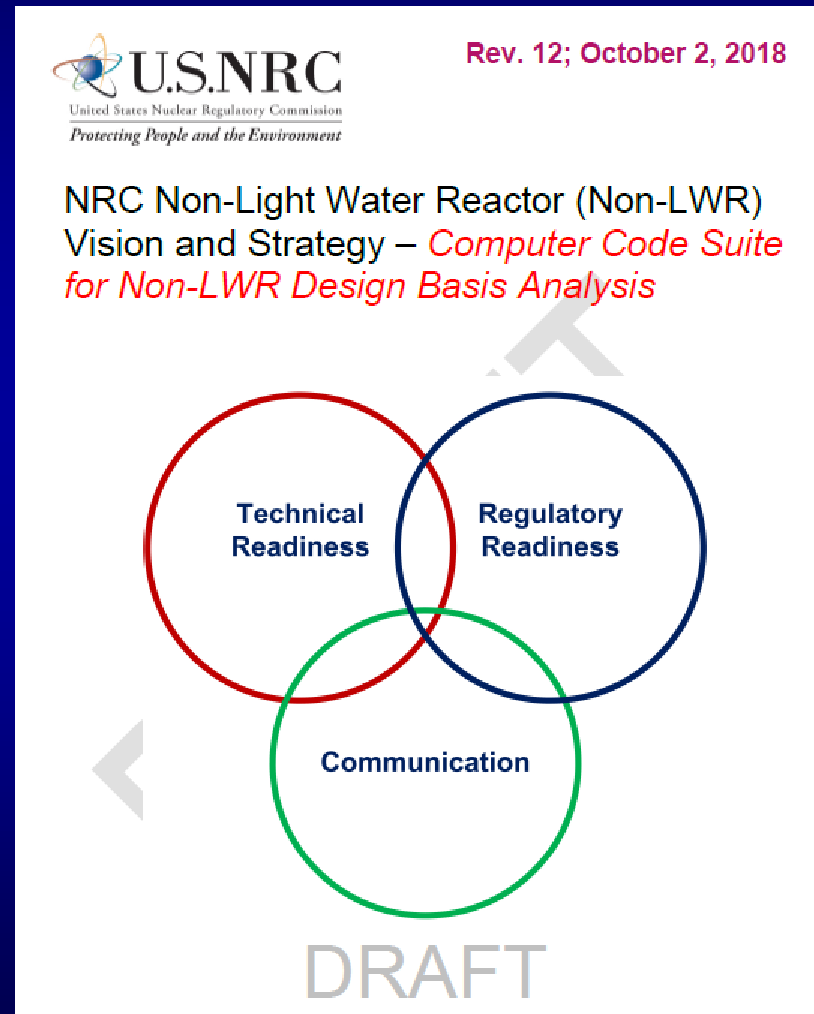
- Cost avoidance. An objective is to minimize cost to the NRC by leveraging DOE tools and influencing development plans.

Codes selected for CRAB satisfy these criteria.

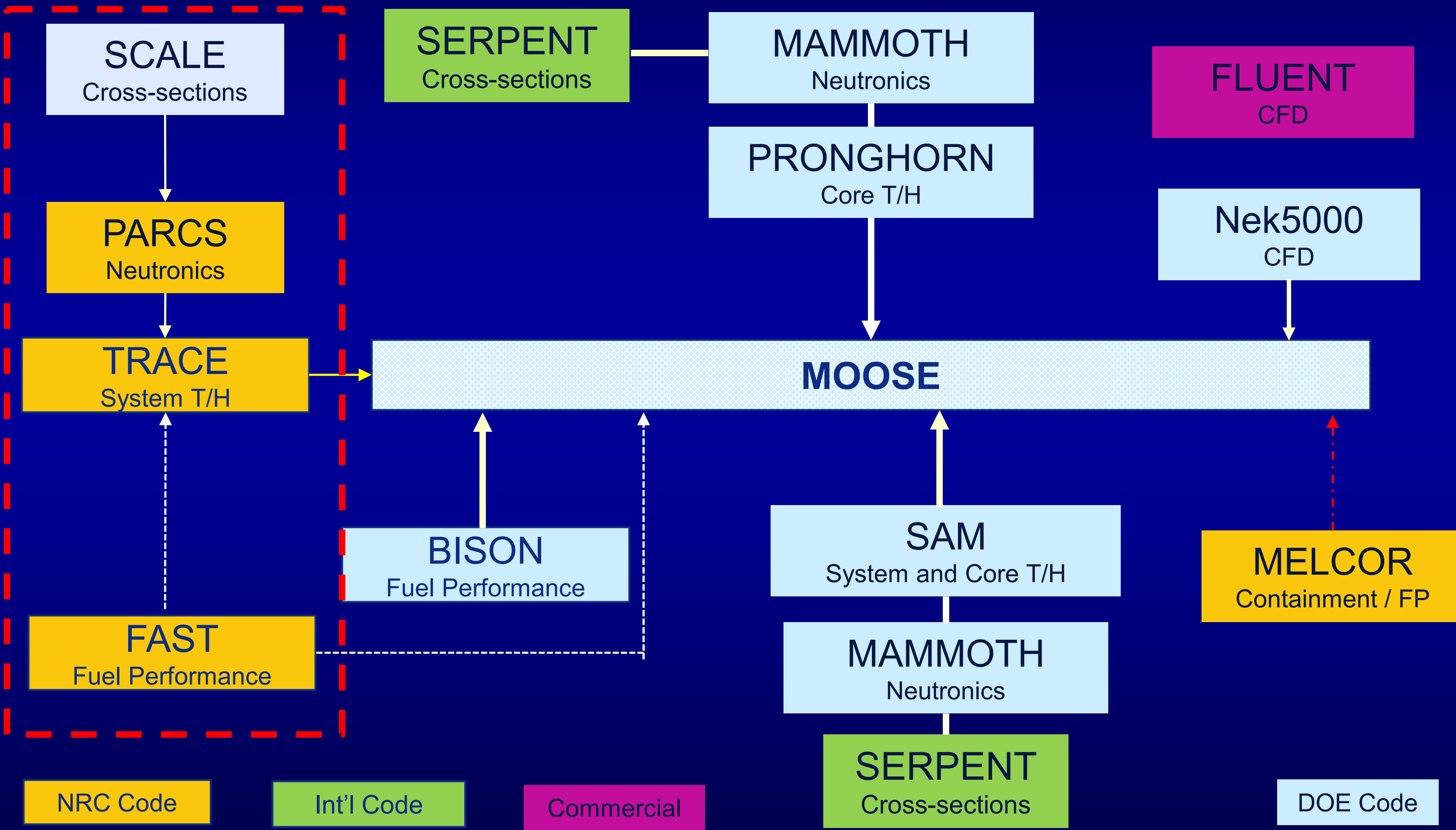
DBE Analysis Codes

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

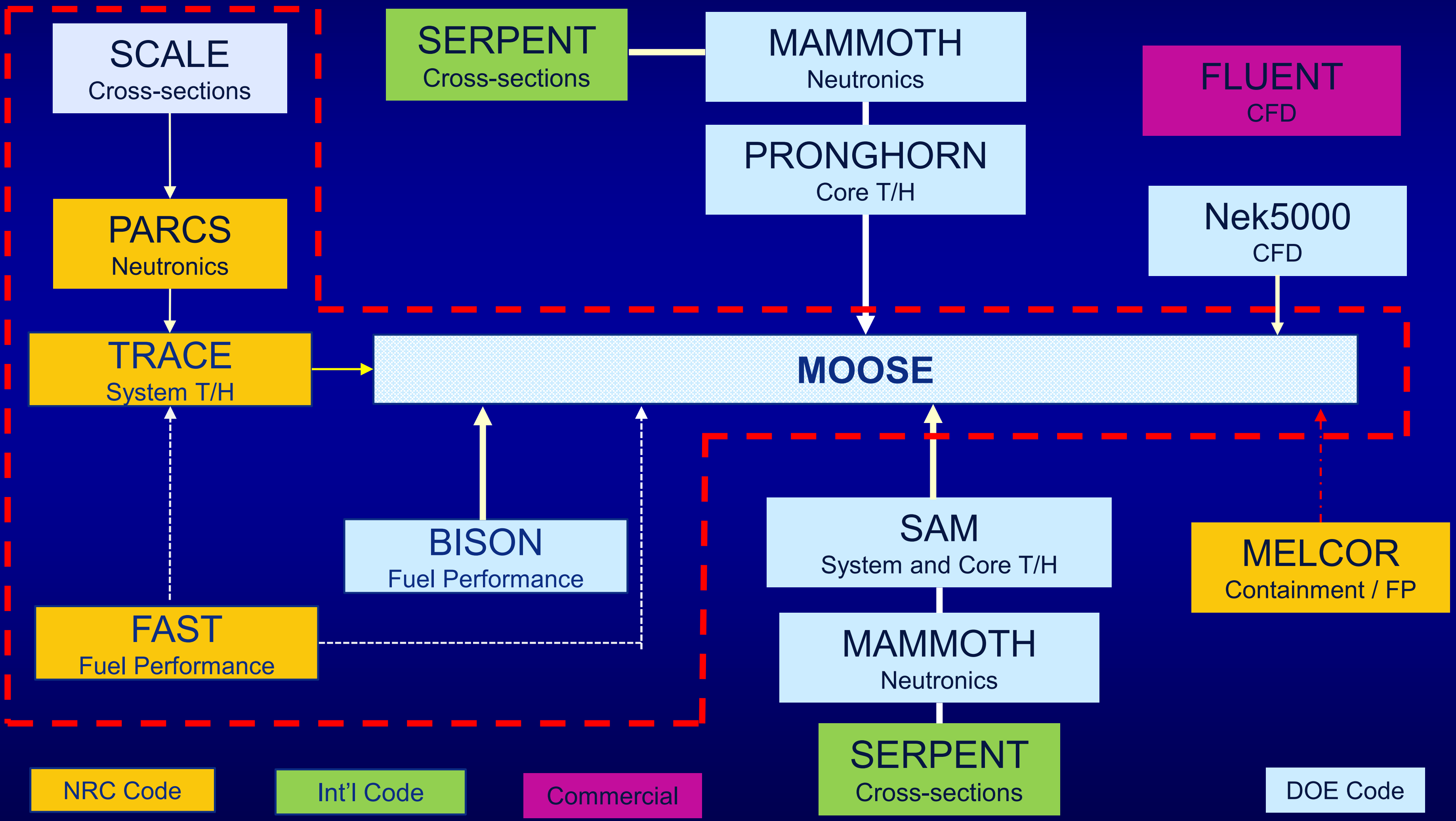
- Reference plant models being developed.



Comprehensive Reactor Analysis Bundle (CRAB for LWRs)

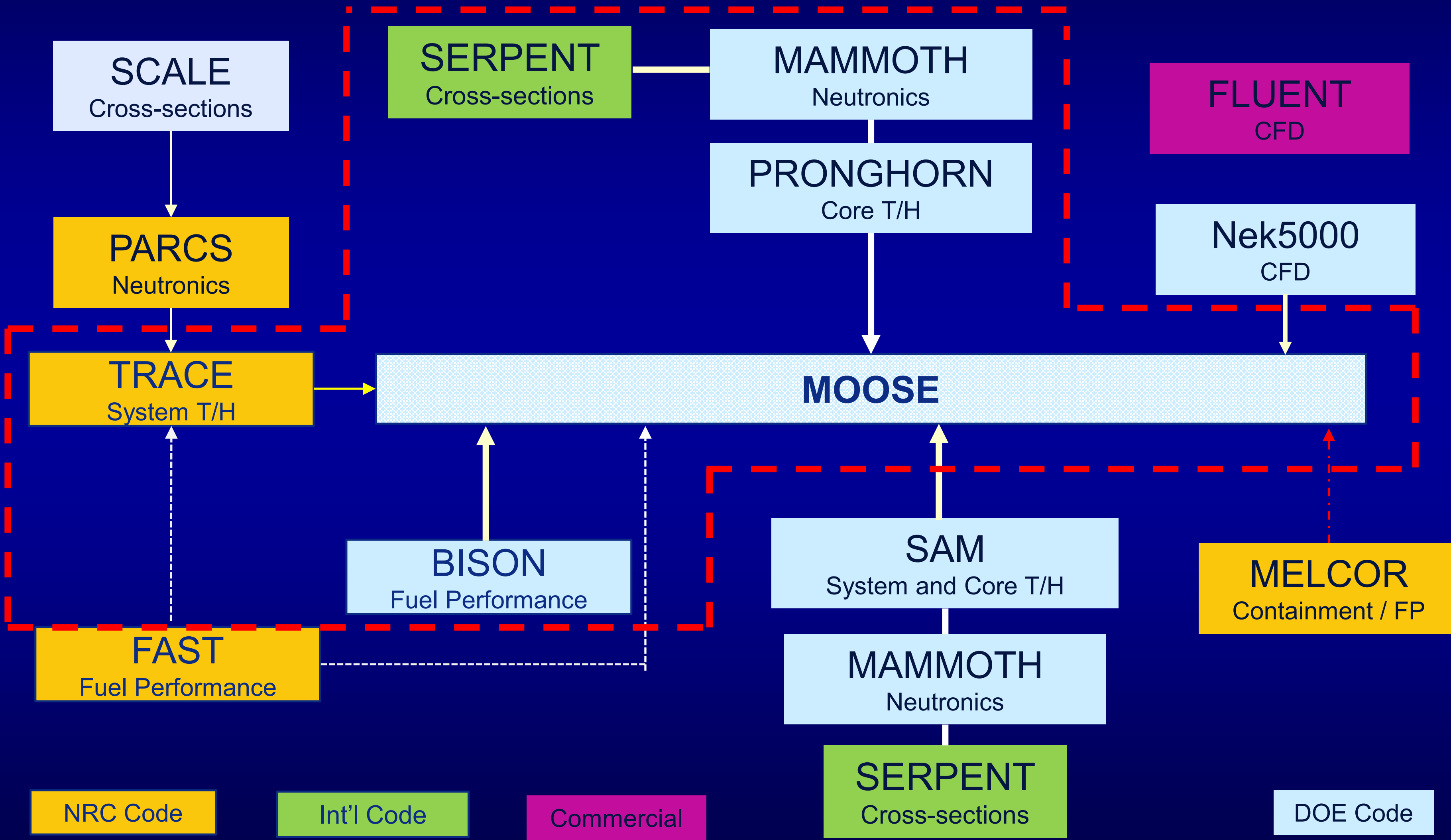


Comprehensive Reactor Analysis Bundle (CRAB for LWRs w/ATF)

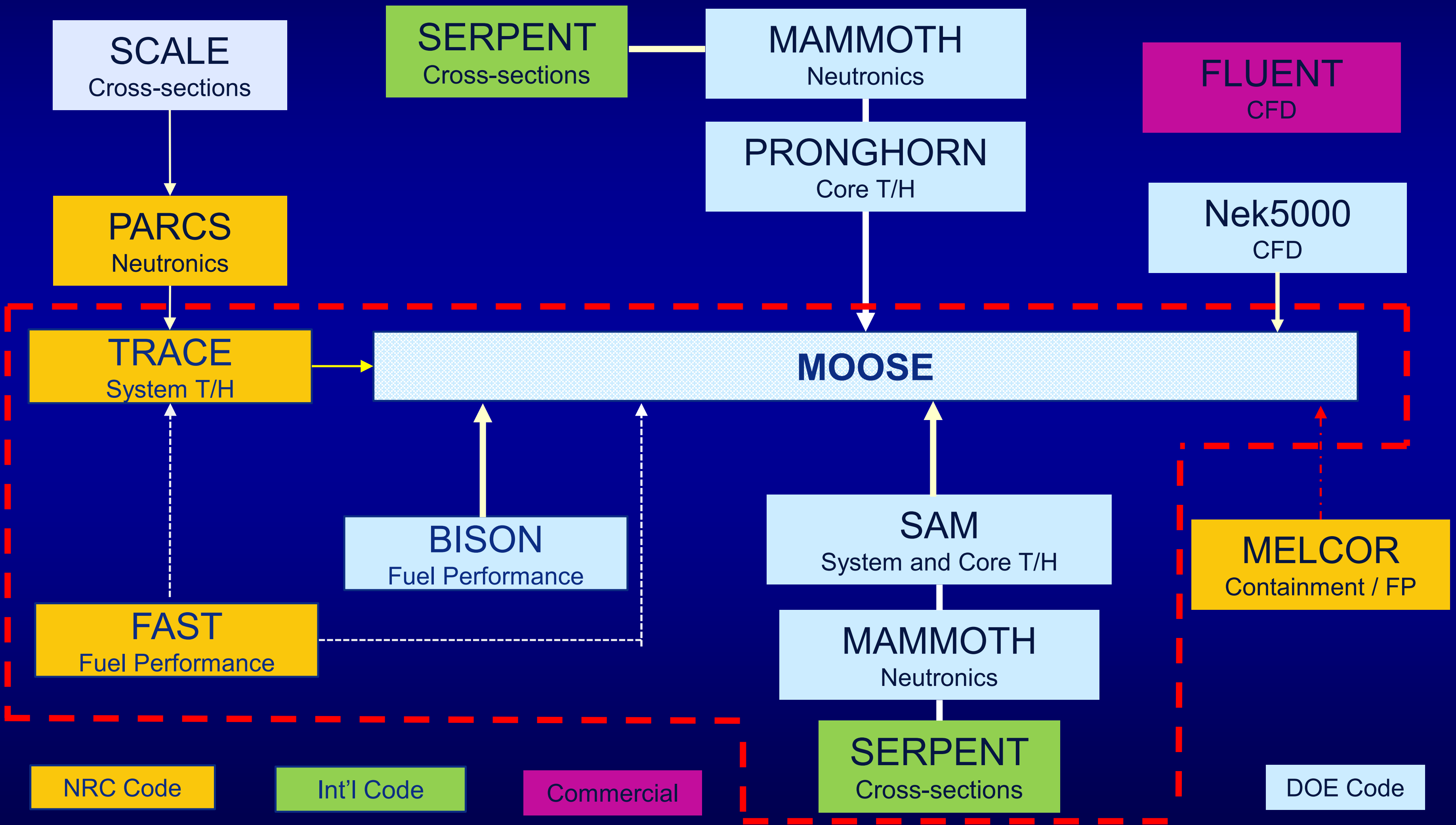


U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Comprehensive Reactor Analysis Bundle (CRAB for GCRs)



Comprehensive Reactor Analysis Bundle (CRAB for Heat Pipe Reactors)



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"

Table 2: Predictive Capability Maturity Model (PCMM) Matrix

Element \ Maturity	Maturity Level 0	Maturity Level 1	Maturity Level 2	Maturity Level 3
Representation and Geometric Fidelity What features are neglected because of simplifications?	• Judgement only • Little or no representation or geometric fidelity for the system	• Significant simplification of the system	• Limited simplification of major components • Geometry is well defined for major components and some minor components • Some peer review conducted	• Essentially no simplifications made • Geometry of all components represented "as built" • Independent peer review conducted
Physics and Model Fidelity How fundamental are the physics and calibration of the models?	• Judgement only • Model forms are unknown or ad hoc • Few physics informed models • No coupling of models	• Some models and correlations are physics based and calibrated to data • Minimal or ad hoc coupling of models	• Physics based models and correlations for all important processes • Significant calibration using SETs and IETs • Some peer review conducted	• All models and correlations are physics based • Sound physical basis for extrapolation • Full coupling of models • Independent peer review conducted
Code Verification Are software errors and poor quality assurance practices?	• Judgement only • Minimum testing of software elements • Little or no SQA	• Code is managed by SQA procedures • Unit and regression testing performed	• Some algorithms are tested to determine convergence • Some features are tested with benchmark solutions • Some peer review conducted	• All of the important algorithms tested to determine convergence • All features and capabilities tested with rigorous benchmark solutions • Independent peer review conducted
Solution Verification Are numerical errors corrupting the results?	• Judgement only • Numerical errors are unknown or have large effect on results	• Numerical effects are qualitatively estimated • Input/output (I/O) verified only by analysis	• Numerical effects quantitatively estimated to be small • I/O independently verified • Some peer review conducted	• Numerical effects are determined to be small • Important simulations can be independently reproduced • Independent peer review conducted
Model Validation How carefully is the accuracy of the simulation and experimental results assessed?	• Judgement only • Few, if any comparisons to measurements in similar systems or applications	• Quantitative assessment of accuracy not directly relevant • Large or unknown experimental uncertainties	• Quantitative assessment of predictive accuracy for some key figures of merit from SETs and IETs • Experimental uncertainties well characterized • Some peer review conducted	• Quantitative assessment of predictive accuracy for all important figures of merit from SETs and IETs at conditions/geometries directly relevant to the application • Experimental uncertainties well characterized • Independent peer review conducted
Uncertainty Quantification and Sensitivity Analysis How thoroughly are uncertainties and sensitivities characterized?	• Judgement only • Only deterministic analyses conducted • Uncertainties and sensitivities not addressed.	• Aleatory and epistemic (A&E) uncertainties propagated, but without distinction • Informal sensitivity studies only	• A&E uncertainties propagated and identified • Quantitative sensitivity analyses conducted • Numerical propagation errors are estimated • Some strong assumptions made • Some peer review conducted	• A&E uncertainties comprehensively treated and properly interpreted • Comprehensive sensitivity analyses conducted • Numerical propagation demonstrated to be small • No significant assumptions • Independent peer review conducted

25

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 <u>technically superior product</u> than can be attained with further development of the NRC's legacy LWR codes only.
- Using the DOE codes provides a <u>significant benefit in resources & schedule</u> to the NRC. DOE has been cooperative in revising their plans to fit our needs and schedule.

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

MELCOR non-LWR ACTIVITIES

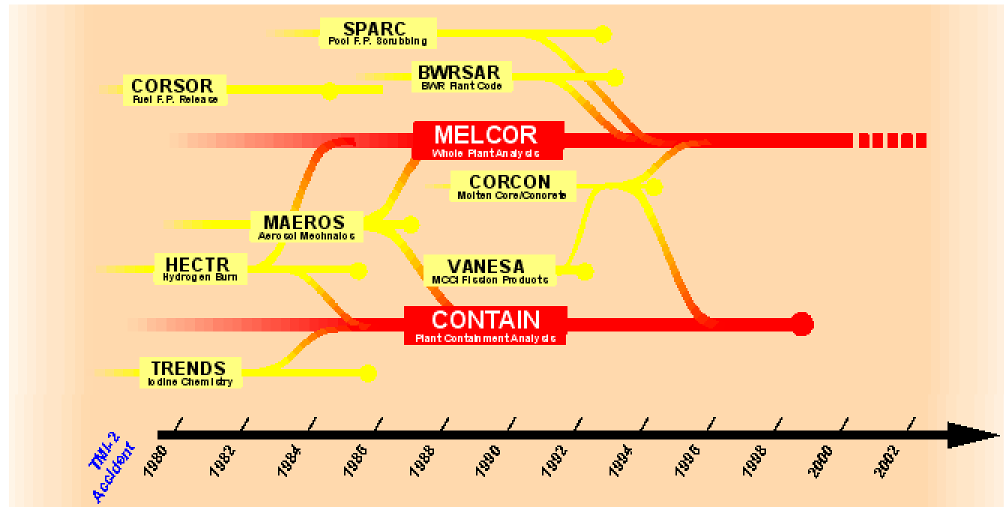
Hossein Esmaili
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

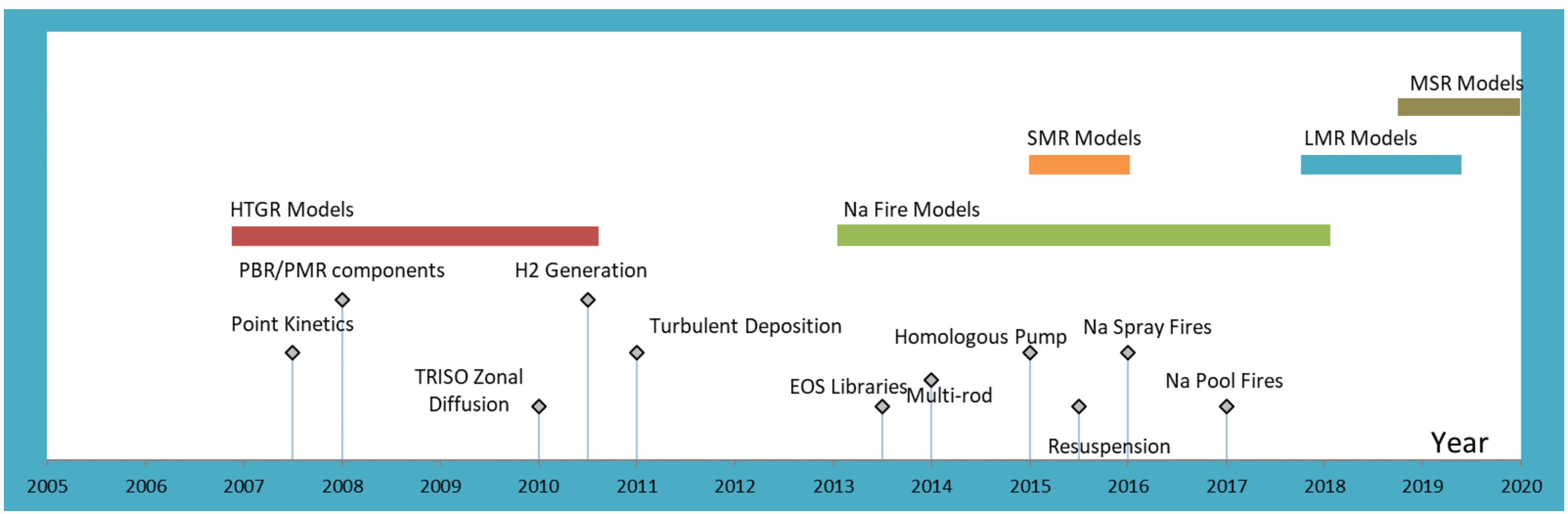
December 13, 2018

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)

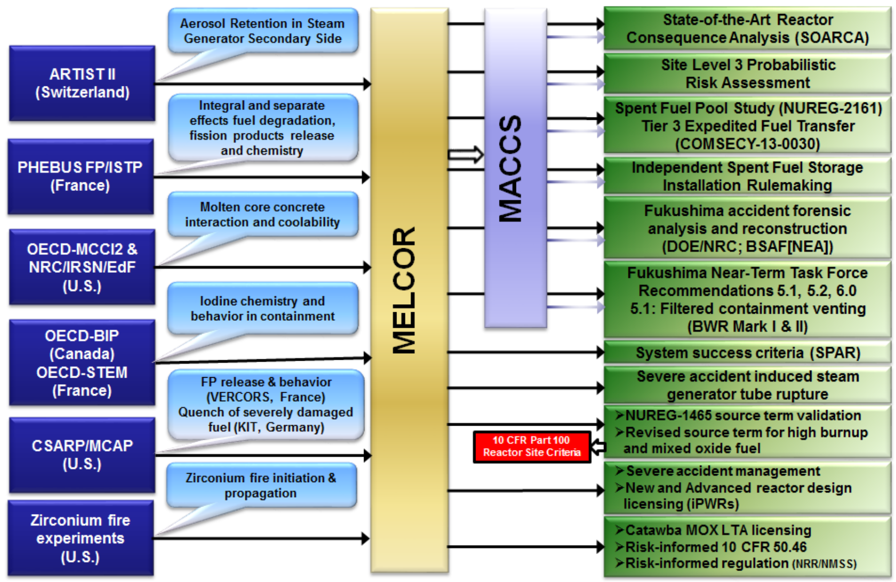


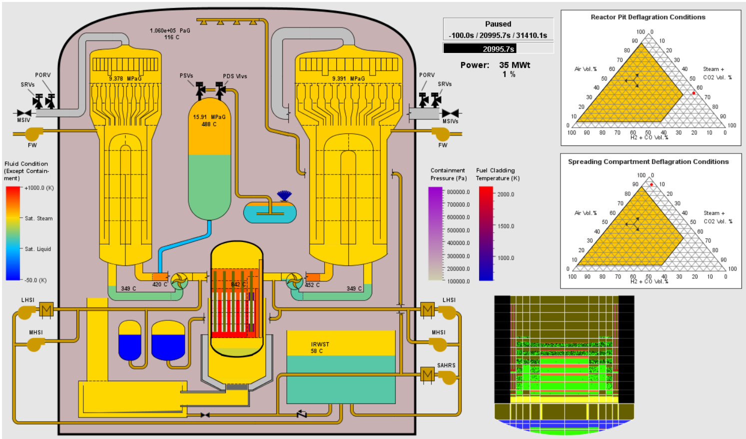


MELCOR Code Development

- 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



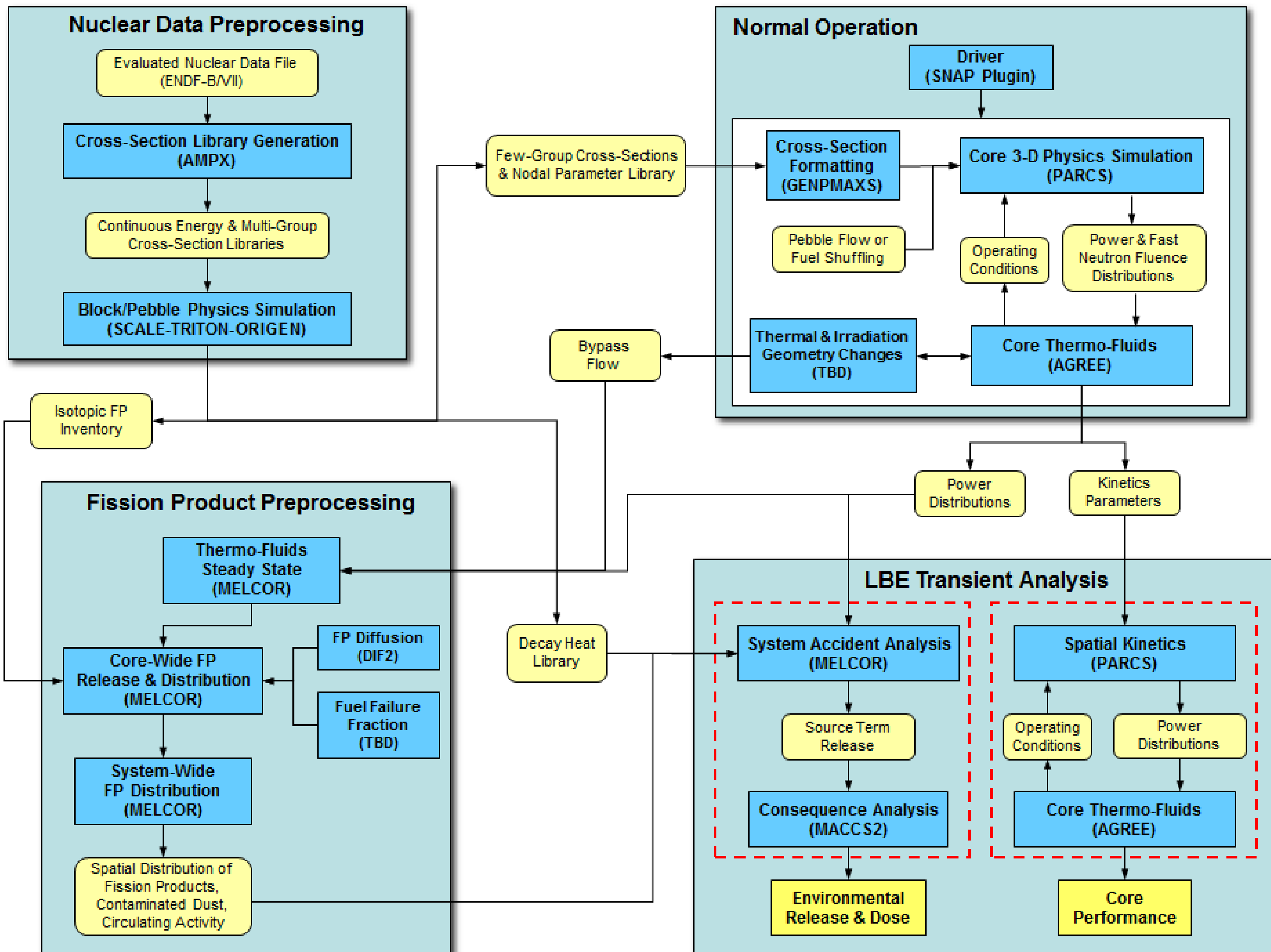


Integrated models required for self-consistent analysis

International Collaboration (CSARP/MCAP/EMUG/AMUG)

Non-LWR Beyond Design Basis Events

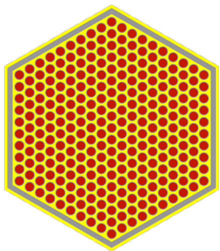
- Development of evaluation models (example HTGR)
  - ACRS Future Plant Designs Subcommittee, April 5, 2011



SCALE Code & Application to MELCOR/MACCS

scale

Nuclear Systems Modeling & Simulation



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

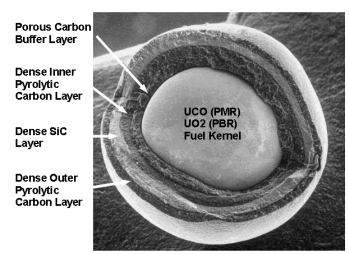
- Oak Ridge Isotope Generation code (ORIGEN)
- Irradiation and decay simulation code
- Fuel depletion and used fuel characterization
- Source terms for accident analyses (operating reactors, spent fuel handling, storage, etc.)
- Structural material activation (in-core, ex-core)
- Material feed and removal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

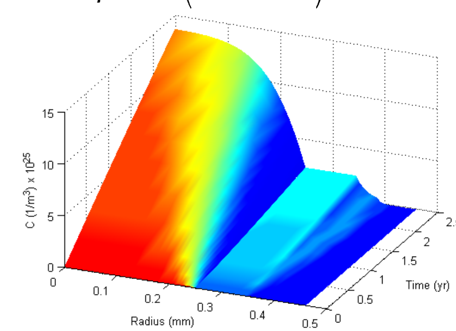
High Temperature Gas Cooled Reactors

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

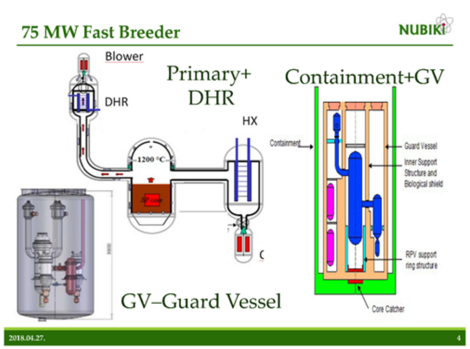
$$\frac{\partial C}{\partial t}=\frac{1}{r^m}\frac{\partial}{\partial r}\left(r^m D\frac{\partial C}{\partial r}\right)-\lambda C+S$$

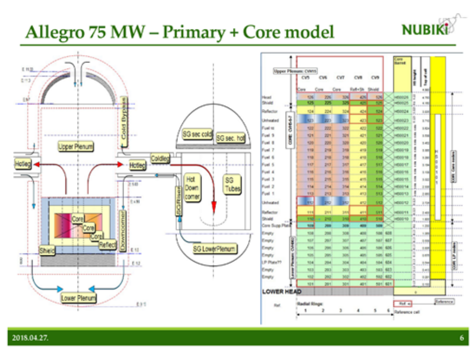


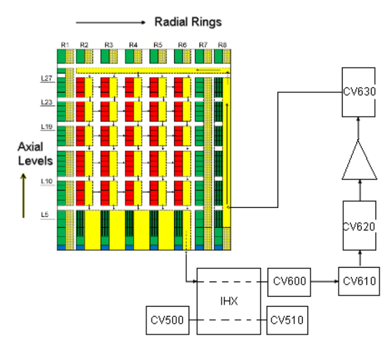


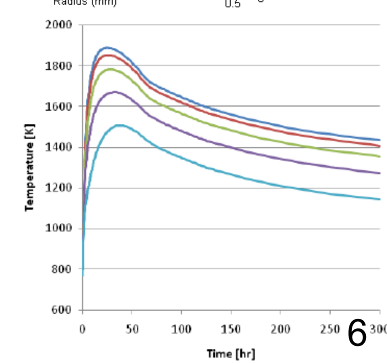
Modeling Gaps

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



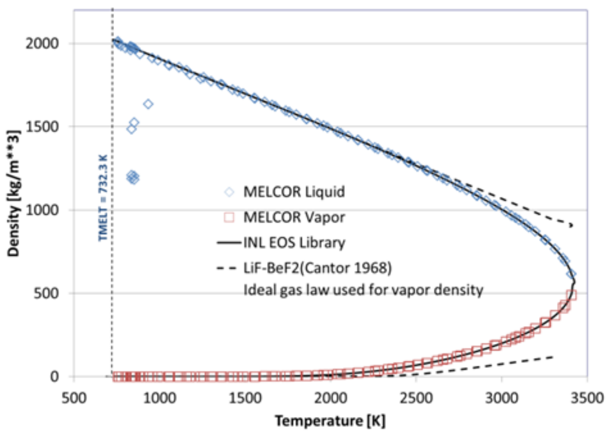


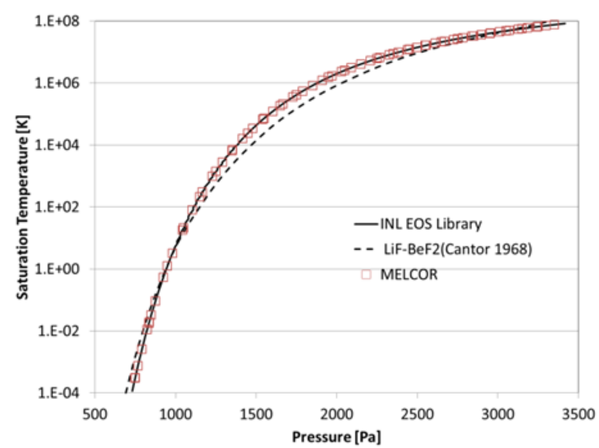




Molten Salt Reactors

- 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.

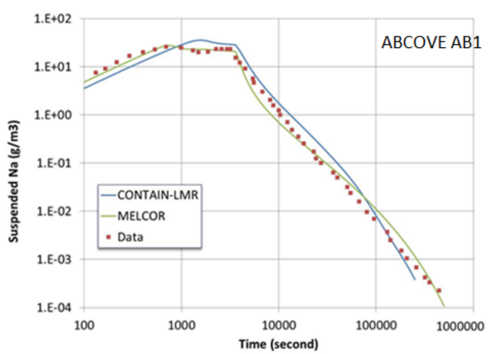




Sodium Fast Reactors

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)

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

Experimental Basis ← PIRT process

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

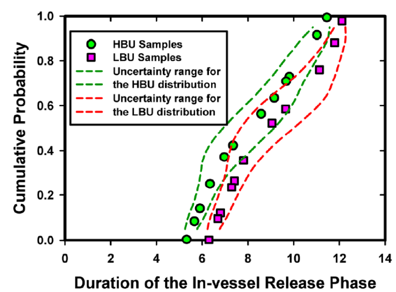
MELCOR

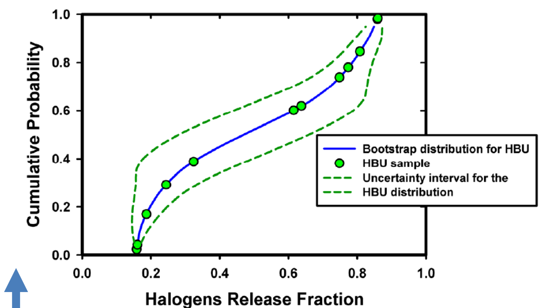
Accident Analysis

Scenario # 1	Scenario # 2
....	....
Scenario # n-1	Scenario # n

Synthesize timings and release fractions

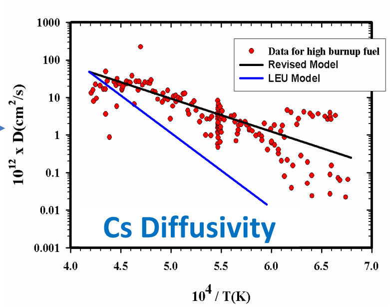
Design Basis Source Term





Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	Late 10 hr
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0

Cesium release from high burnup fuel
Comparison to results of VERCORS Test



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

Powers, et al. "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup or MOX Fuel", SAND2011-0128 January 2011

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

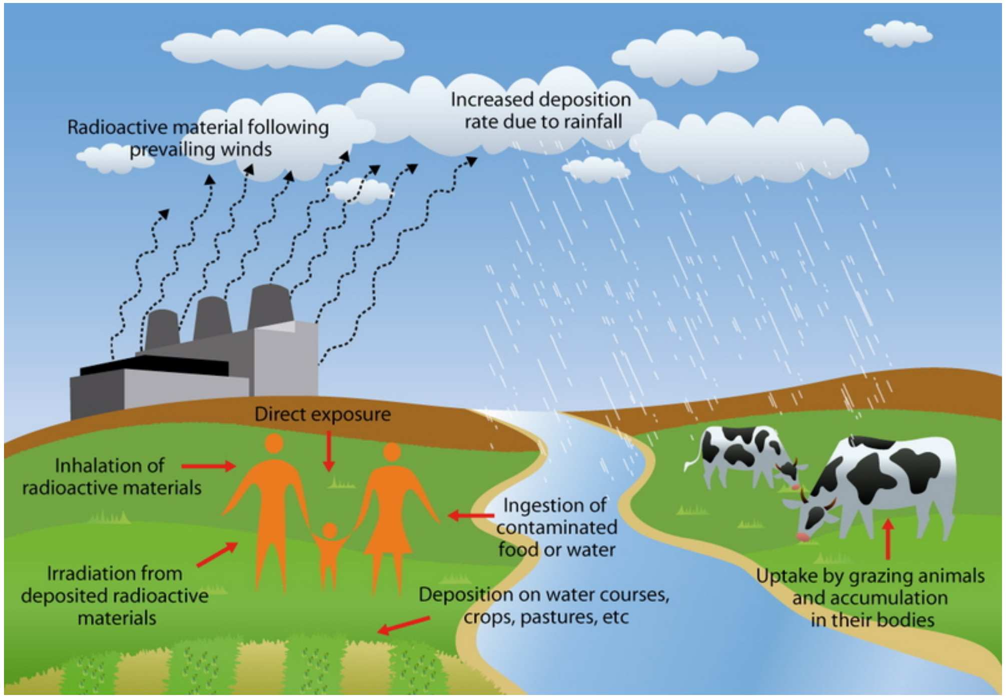
Consequence Analysis (MACCS) Code Development Plan for Non-LWRs

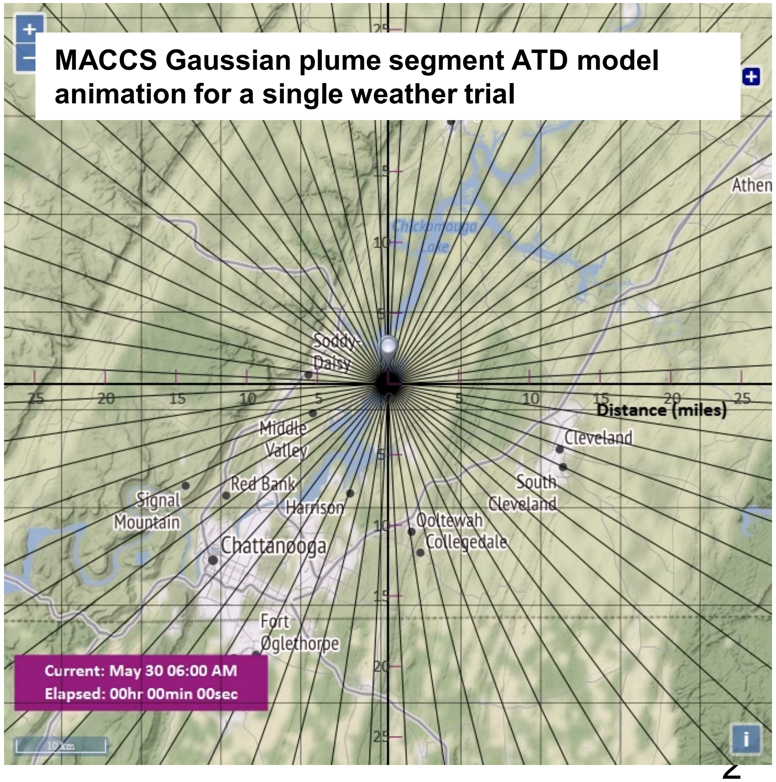
Jonathan Barr
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

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

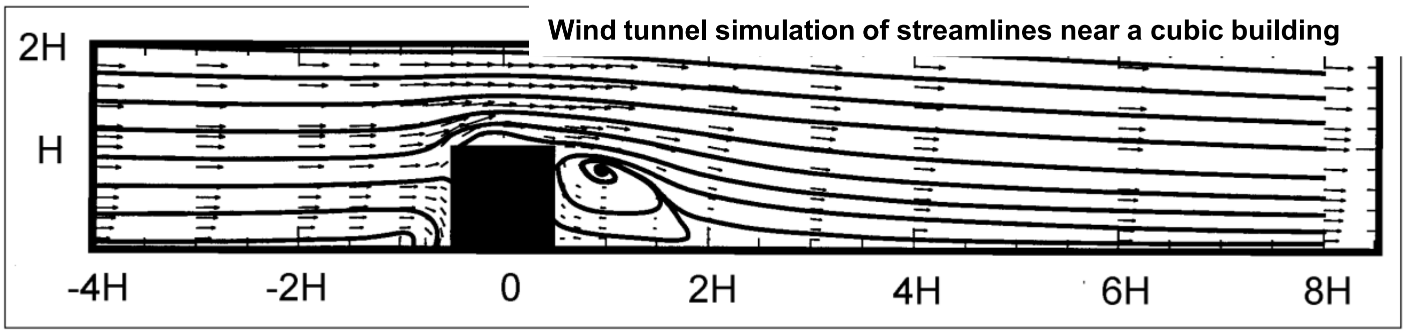
- Code development plans for design-specific issues
  - Radionuclide screening
  - Radionuclide size
  - Radionuclide chemical form
  - Radionuclide shape factor
  - Tritium

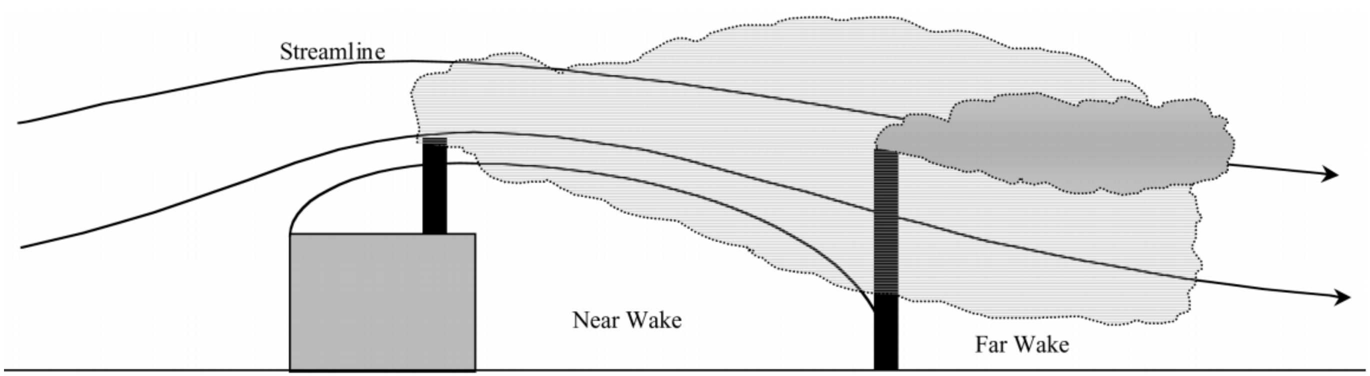
- Code development plans for site-related issues
  - Near-field atmospheric transport
  - Decontamination modeling

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

Wind tunnel simulation of streamlines near a cubic building





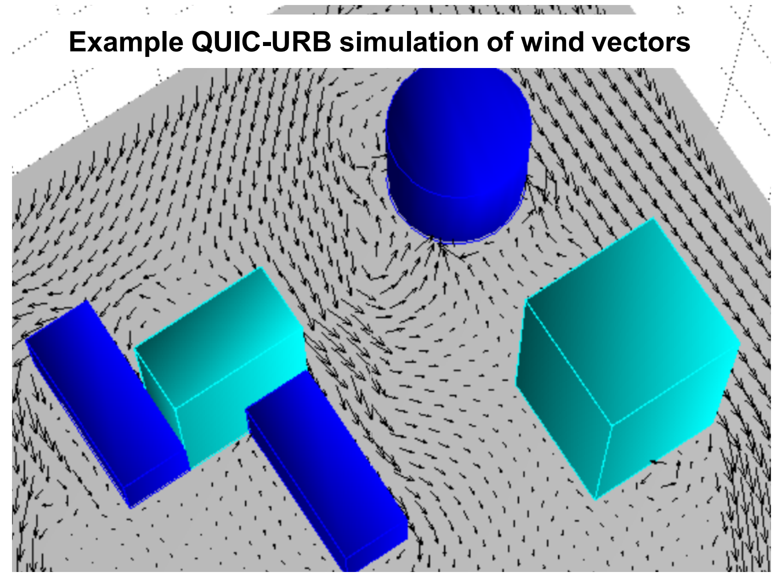
Lloyd L. Schulman , David G. Strimaitis & Joseph S. Scire (2000) Development and Evaluation of the PRIME Plume Rise and Building Downwash Model, Journal of the Air & Waste Management Association, 50:3, 378-390

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

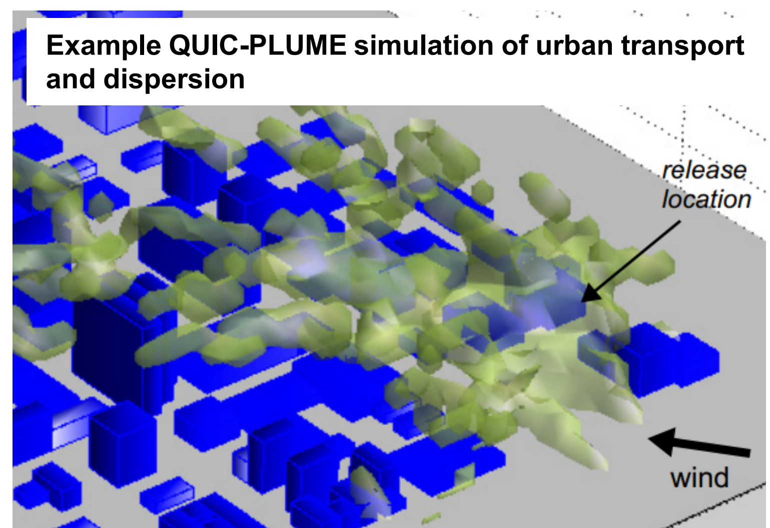
Near-Field Atmospheric Transport

- 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

QUIC Factsheet, Los Alamos National Laboratory

NUCLEAR INNOVATION ALLIANCE

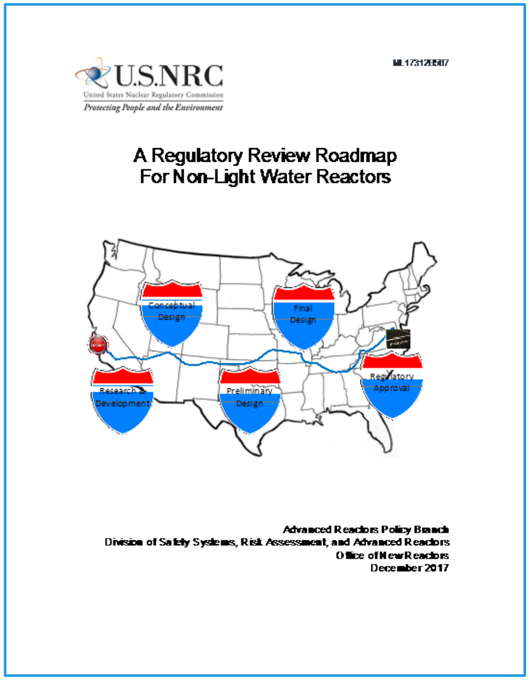
Establishing Interface Requirements in Support of Staged Licensing

December 13, 2018

Ashley Finan
ashley@nuclearinnovationalliance.org

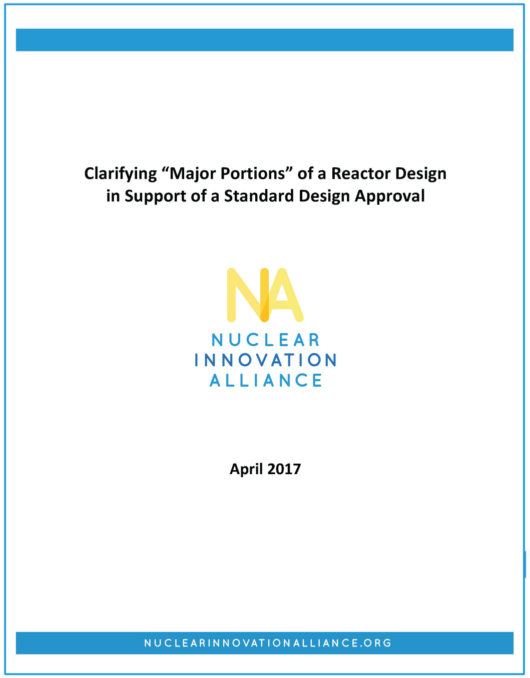
Background Documents

- 10 CFR Part 52, Subpart E allows an applicant to seek standard design approval for either an entire plant or “major portions” thereof
- NRC document: “A Regulatory Review Roadmap for Non-Light Water Reactors” (ML17312B567)
- NIA report: “Clarifying ‘Major Portions’ of a Reactor Design in Support of a Standard Design Approval” (ML17128A507)
- NRC staff provided feedback on this report on July 20, 2017 (ML17201Q109)


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

A Regulatory Review Roadmap
For Non-Light Water Reactors

Advanced Reactors Policy Branch
Division of Safety Systems, Risk Assessment, and Advanced Reactors
Office of New Reactors
December 2017


Clarifying “Major Portions” of a Reactor Design in Support of a Standard Design Approval

NUCLEAR INNOVATION ALLIANCE

April 2017

NUCLEARINNOVATIONALLIANCE.ORG

NIA Draft Report: “Establishing Interface Requirements in Support of Staged Licensing”

Table of Contents:

NUCLEAR INNOVATION ALLIANCE

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

NUCLEAR INNOVATION ALLIANCE

Figure 1: Current Project Risk/Investment Profile Relative to Detailed Design & Licensing

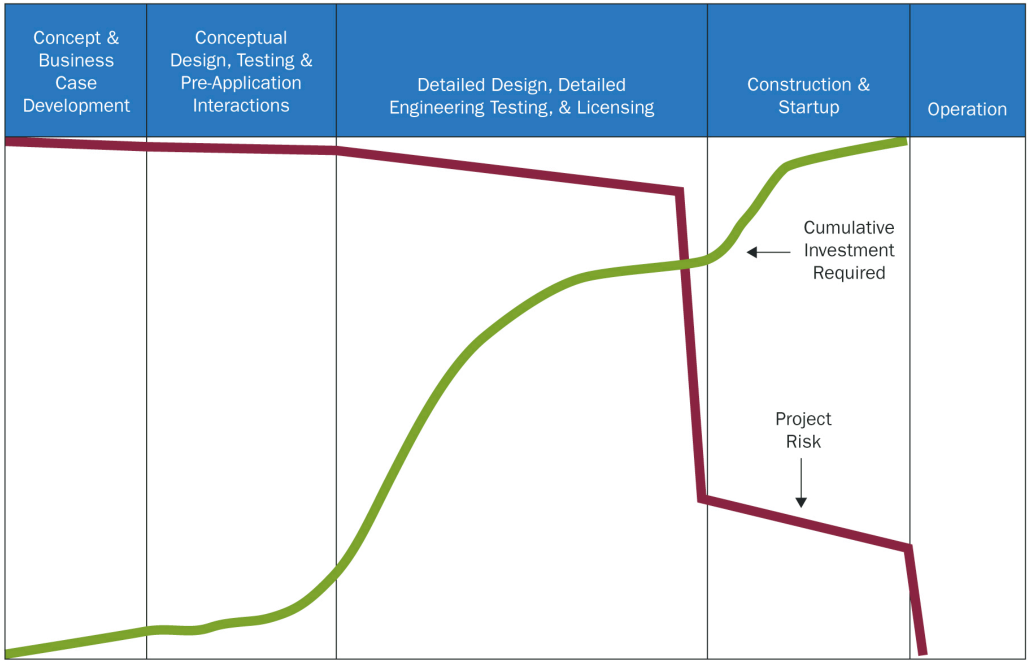
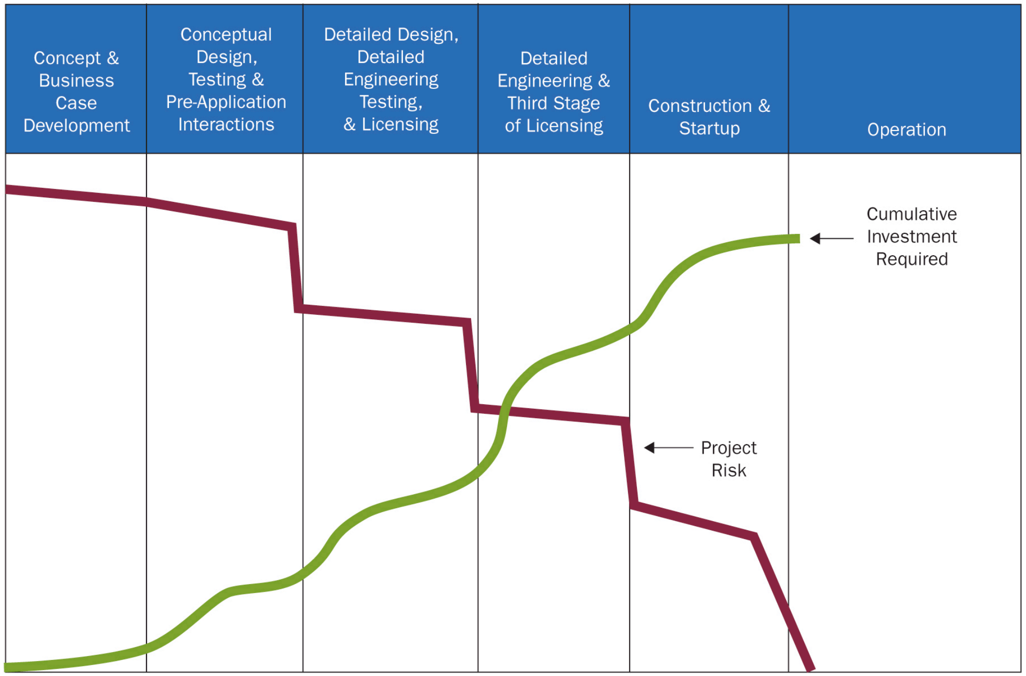


Figure 2: Desirable Project Risk/Investment Profile Relative to Detailed Design & Licensing



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

NUCLEAR INNOVATION ALLIANCE

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

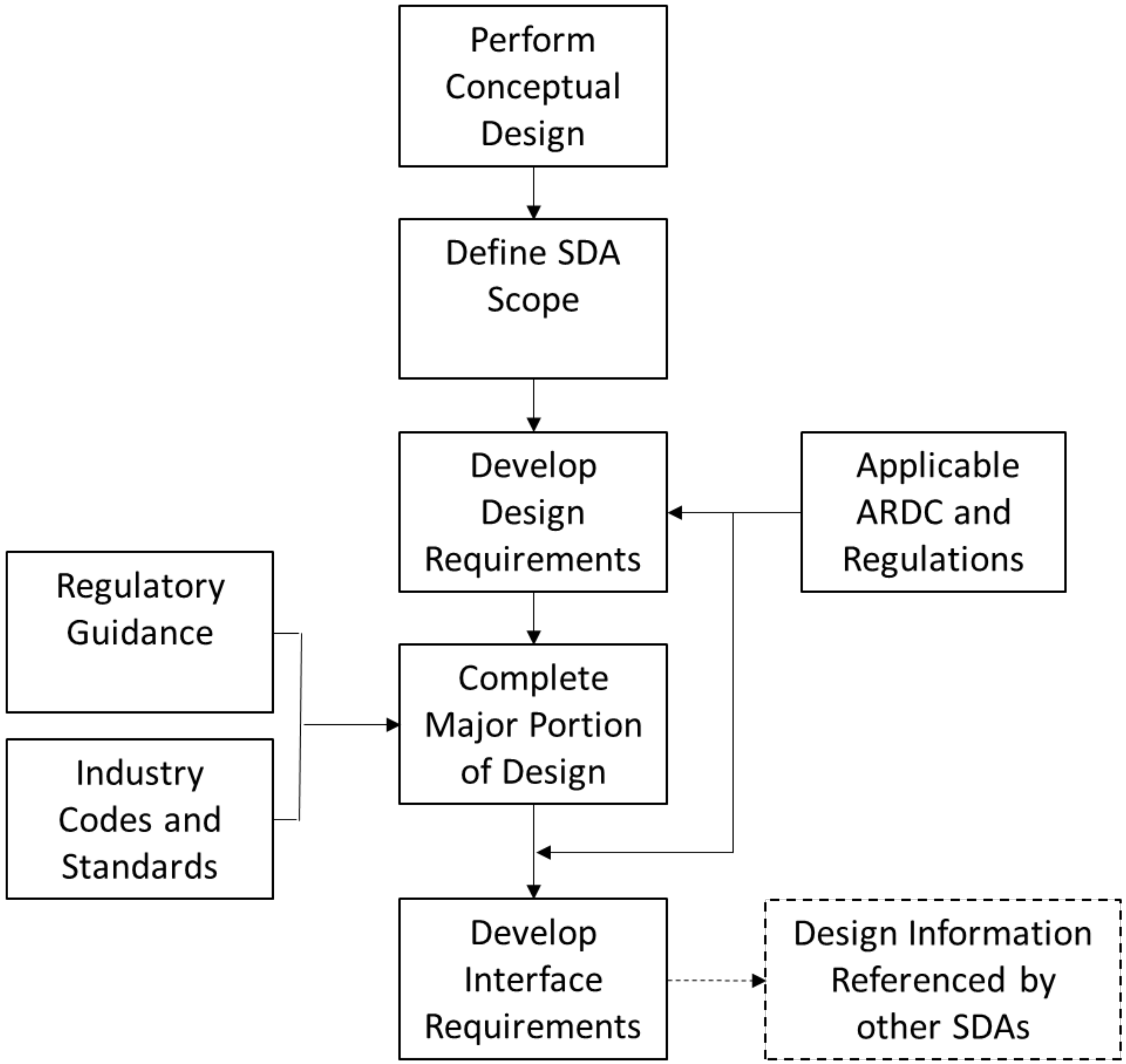
NUCLEAR INNOVATION ALLIANCE

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

NUCLEAR INNOVATION ALLIANCE

Process for Developing Interface Requirements in Support of an SDA



NUCLEAR INNOVATION ALLIANCE

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

NUCLEAR INNOVATION ALLIANCE

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

NUCLEAR INNOVATION ALLIANCE

ARDC	Title	Sample Interface Requirements for RVAC System
2	Design basis for protection against natural phenomena	Interface Requirement The ability of the SSCs of the RVAC to withstand the design basis natural phenomena will be addressed in the FSAR. The comparison of the FSAR design assumptions to those relating to an actual site will be addressed in a future submittal. Adequate margin should be included in the assumed values for the natural phenomena to provide flexibility in siting the design. The FSAR will specify seismic, hurricane, and tornado design parameters (e.g., earthquake design response spectra, soil conditions, tornado and hurricane wind speeds, etc.). These parameters will be compared to those evaluated for a future site.
3	Fire protection	Interface Requirement The RVAC is required to have a fire protection program. The fire protection program will be addressed in a future submittal. The FSAR will include a commitment that the materials used in the RVAC structure will use noncombustible and fire- resistant materials wherever practical, particularly in locations with SSCs important to safety.

Next Steps

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

Thank you!

NUCLEAR INNOVATION ALLIANCE

Thank you

Feedback & Questions

Please feel welcome to send additional input at any time to Ashley Finan (ashley@nuclearinnovationalliance.org).

NUCLEAR INNOVATION ALLIANCE

United States
NIC
Nuclear Industry Council

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

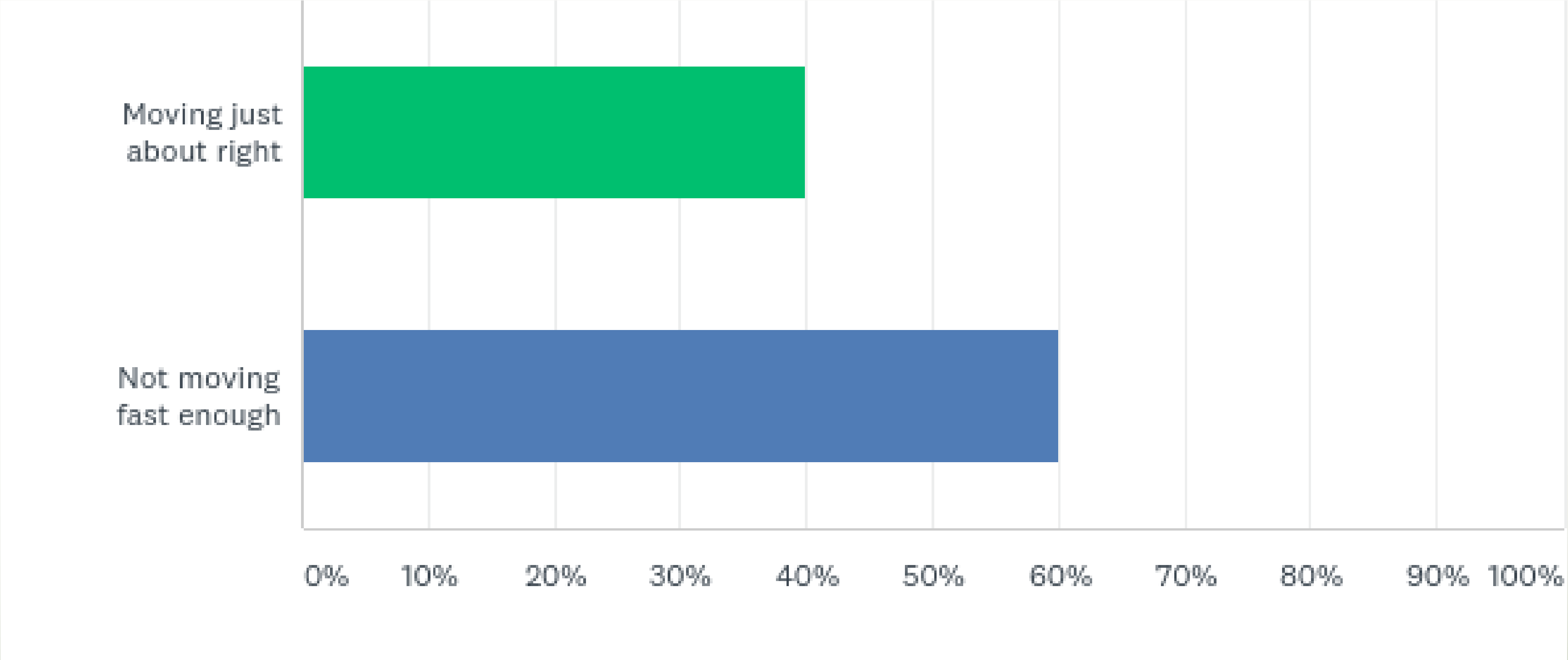
- USNIC conducted a third in a series survey of 16 leading U.S. Advanced Reactor technology developers with regard to DOE Initiatives
- 15 Developers responded, one respondent per company
- This was a blind survey so individual results were not identified

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

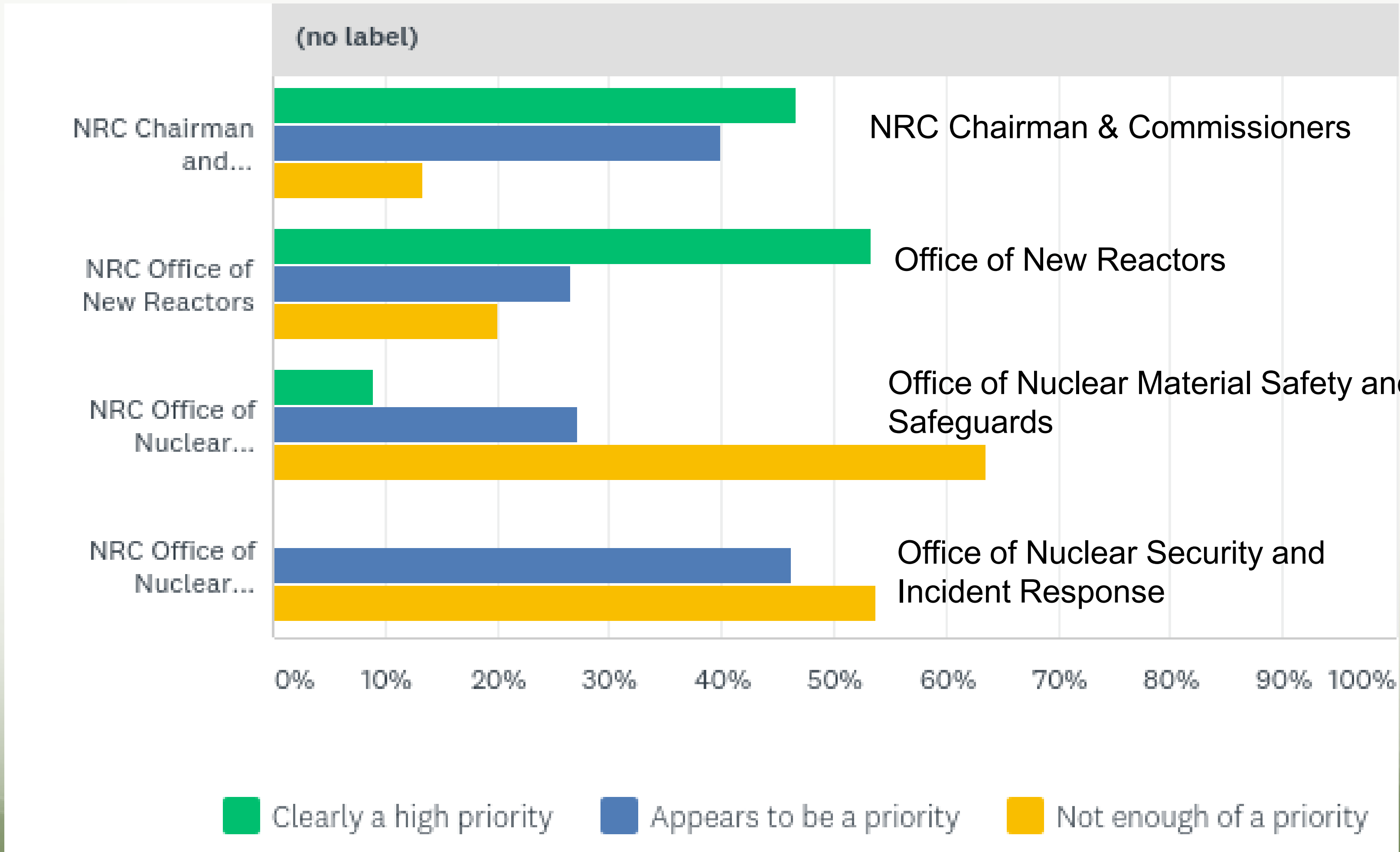
United States NIC Nuclear Industry Council

Q1: Pace of the NRC's Advanced Reactor Licensing Transformation: Rate the pace of the NRC's Preparation Activities for Advanced Reactor licensing?

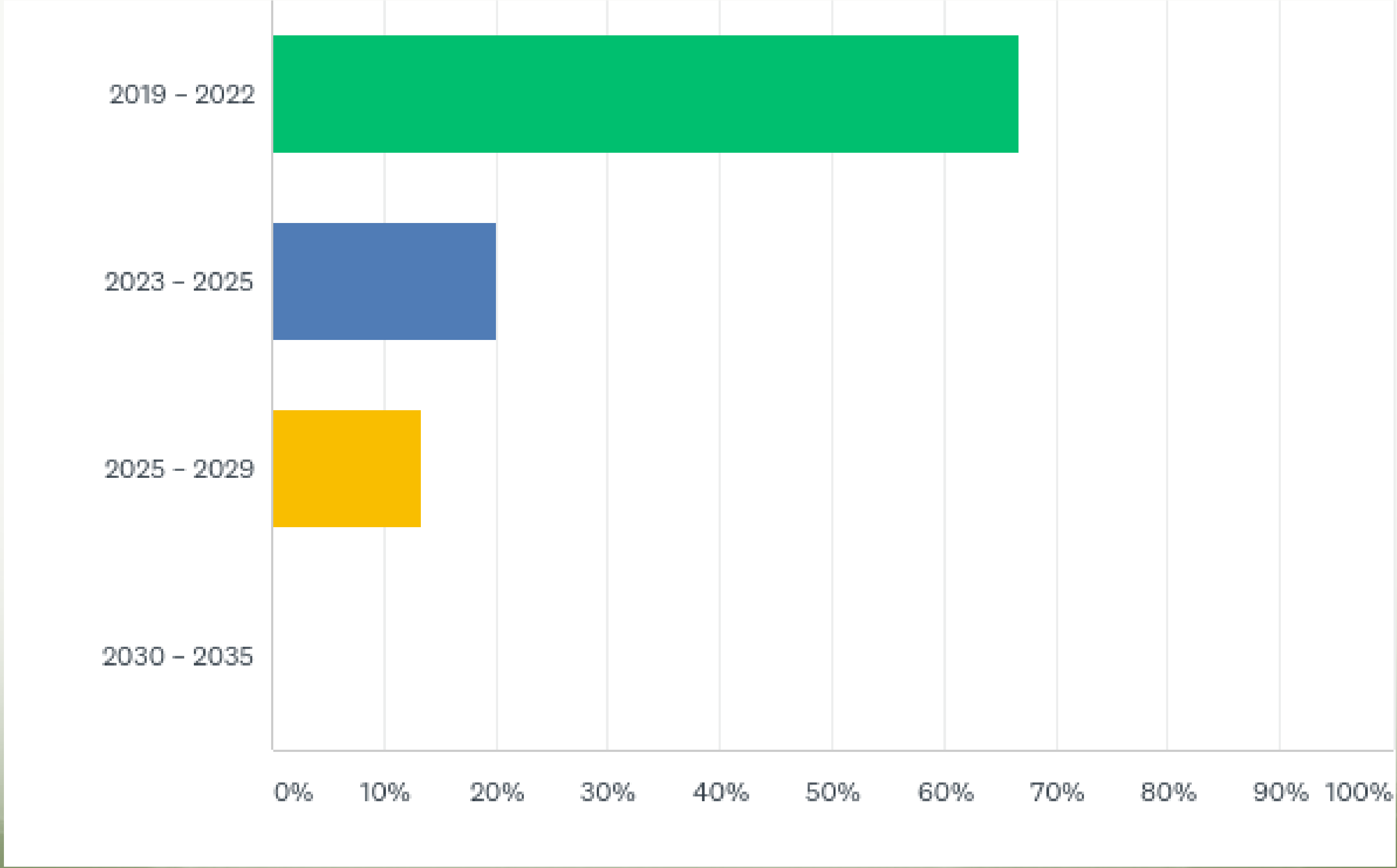


United States NIC Nuclear Industry Council

Q2: NRC Support for Advanced Reactor Licensing Transformation: Please rank the NRC Offices' prioritization of Advanced Reactor transformation?

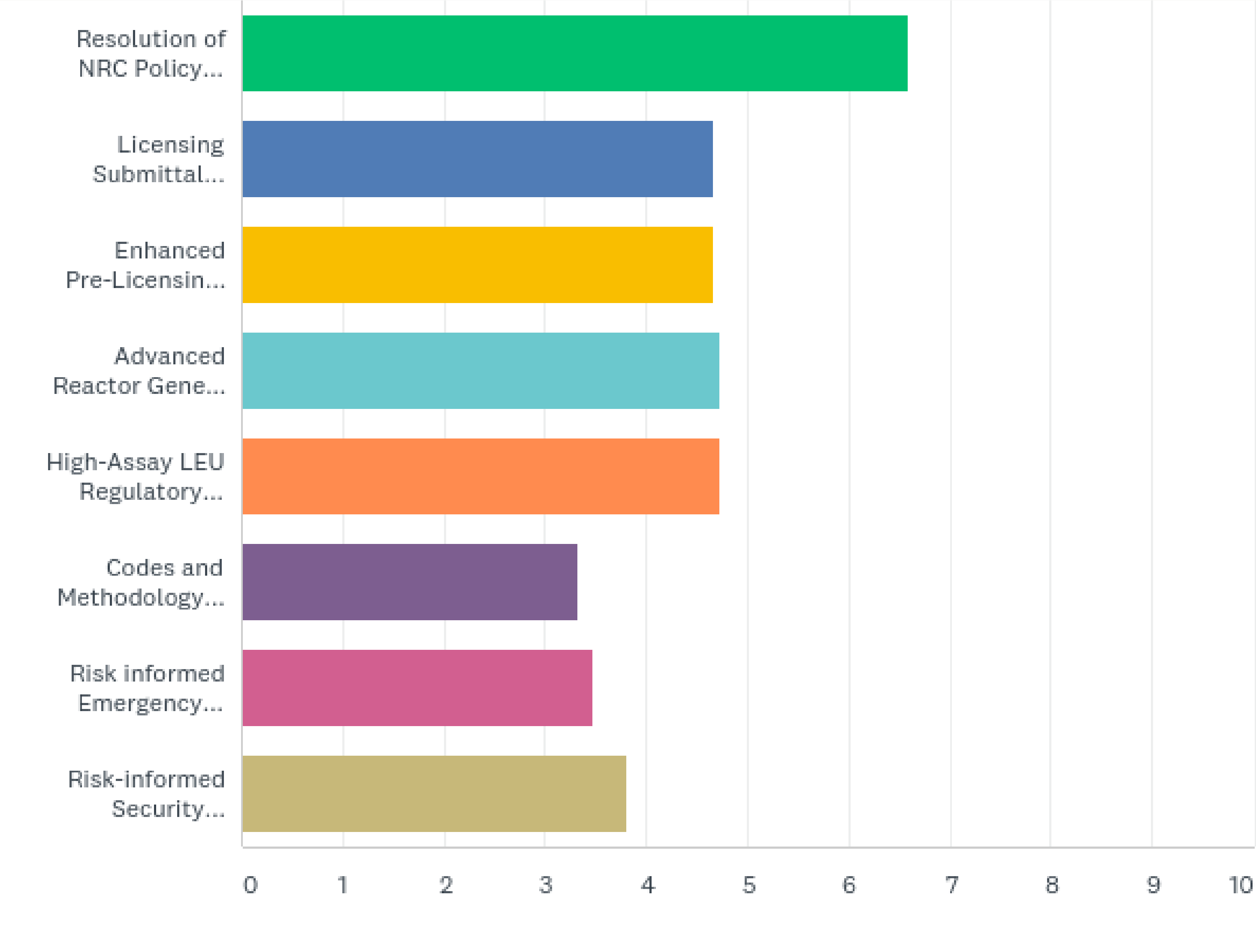


Q3: Planning Timeframe for Licensing Application Submittals: What should the NRC and DOE's Planning Timeframe be for new Advanced Reactor License Applications?



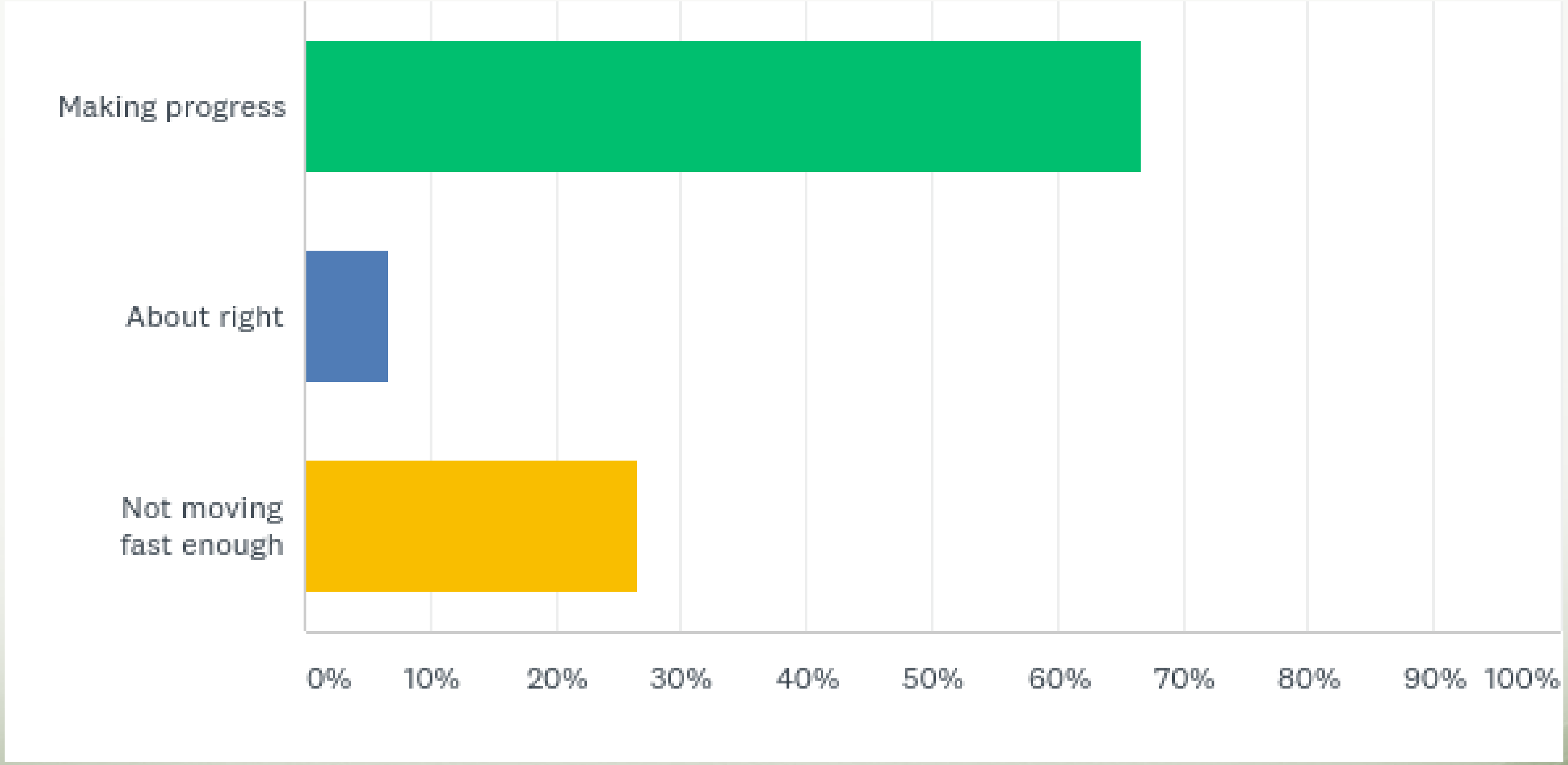
United States NIC Nuclear Industry Council

Q4: Focus for NRC Advanced Reactors Licensing Transformation in 2019: What should the NRC's key Licensing Transformation Focus be in? (ranked)

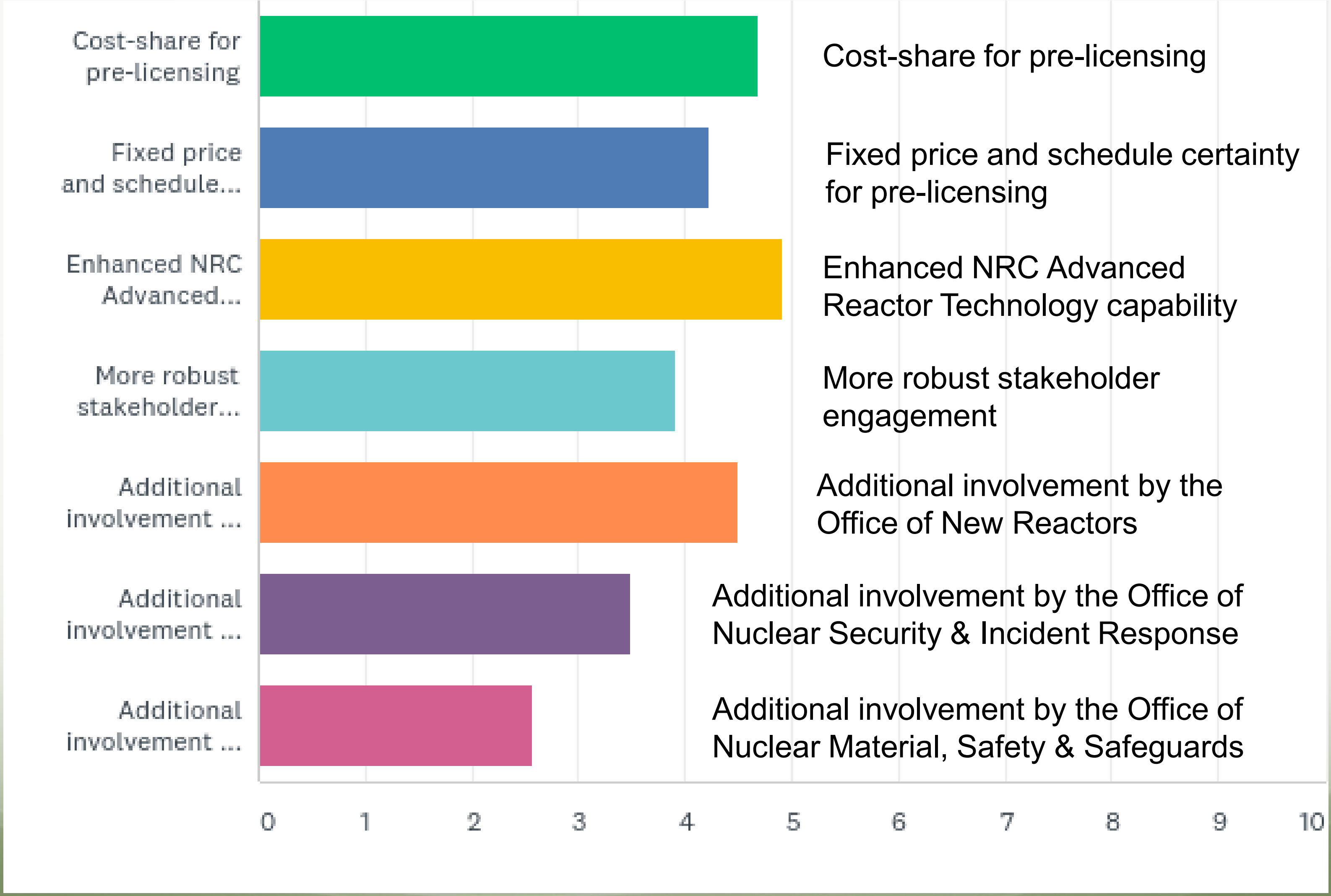


United States NIC Nuclear Industry Council

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?

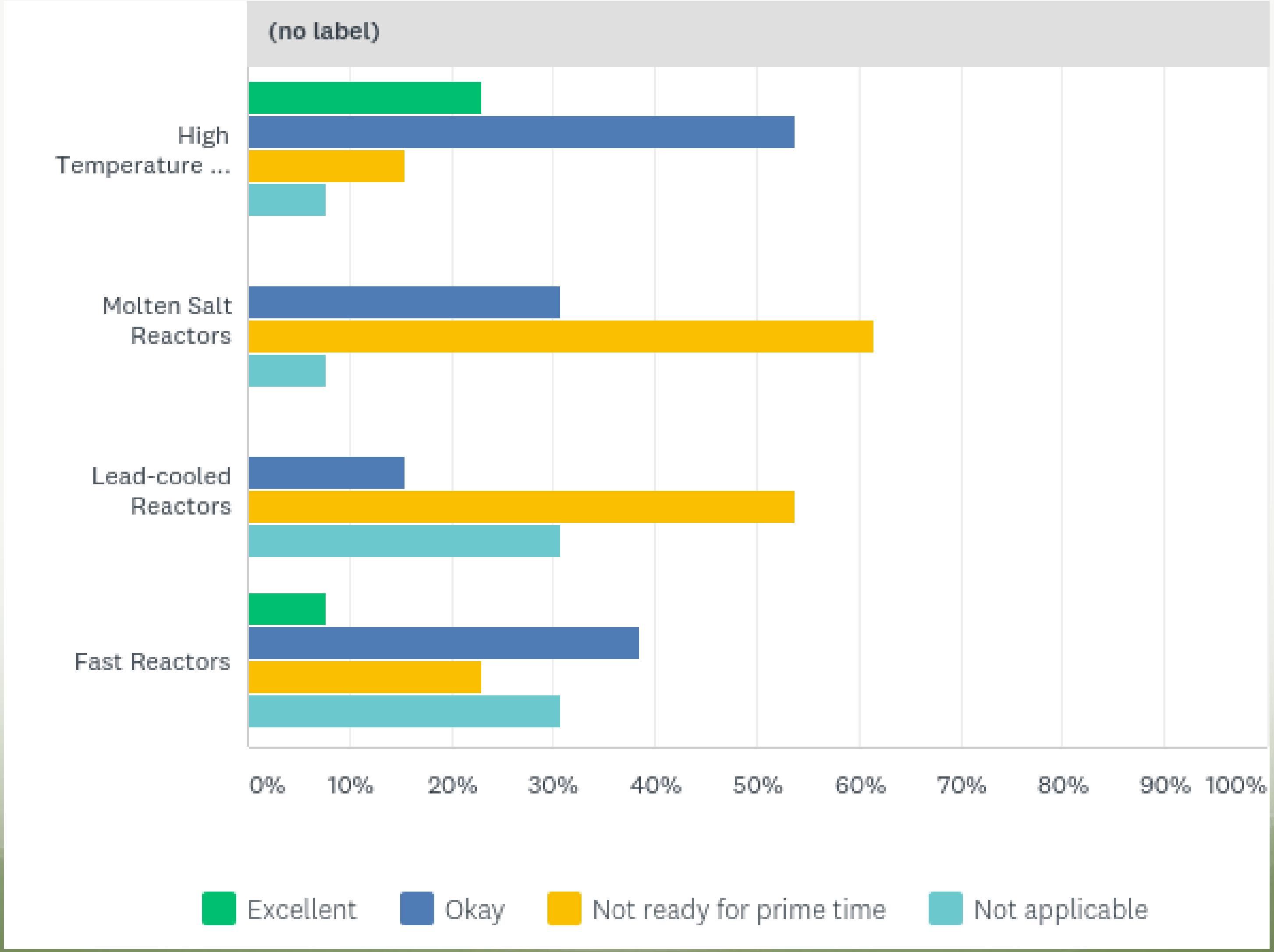


Q6: Enhanced Pre-Licensing Engagement: What actions would most improve the NRC's pre-licensing engagement (rank in order of priority)?



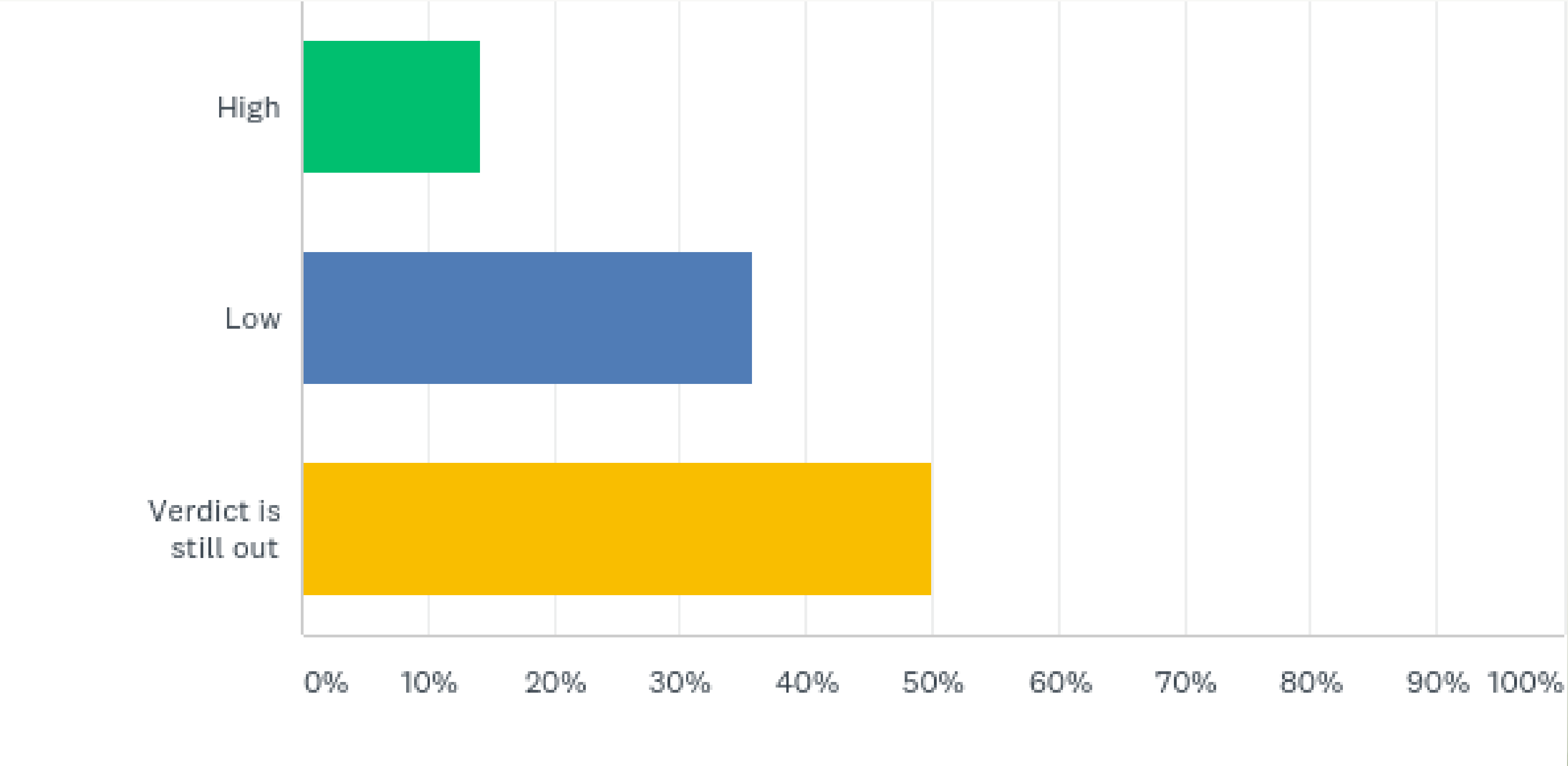
United States NIC Nuclear Industry Council

Q7: NRC Advanced Reactors Technical Capability: Please rate the NRC's Advanced Reactor technology technical capability?

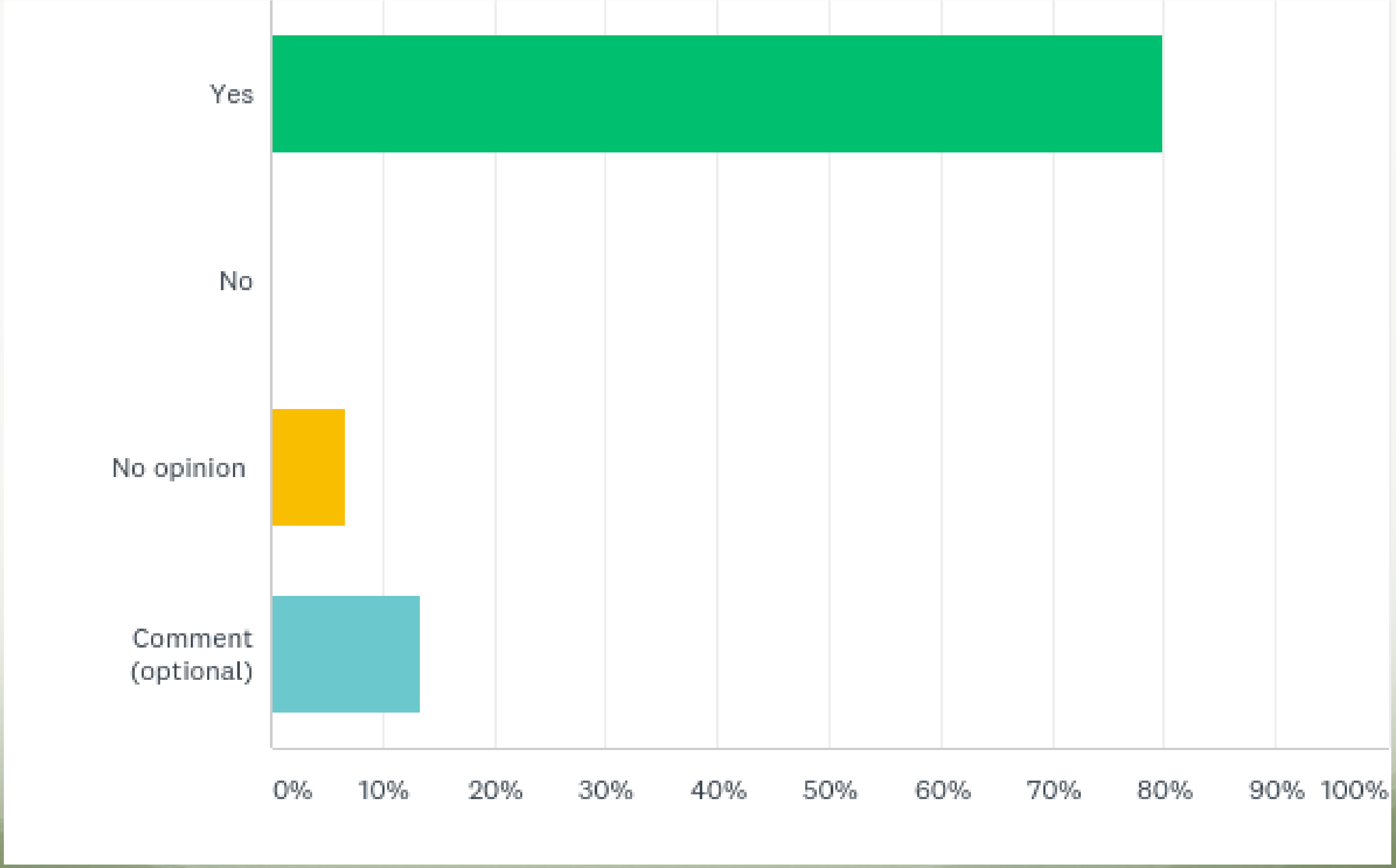


United States NIC Nuclear Industry Council

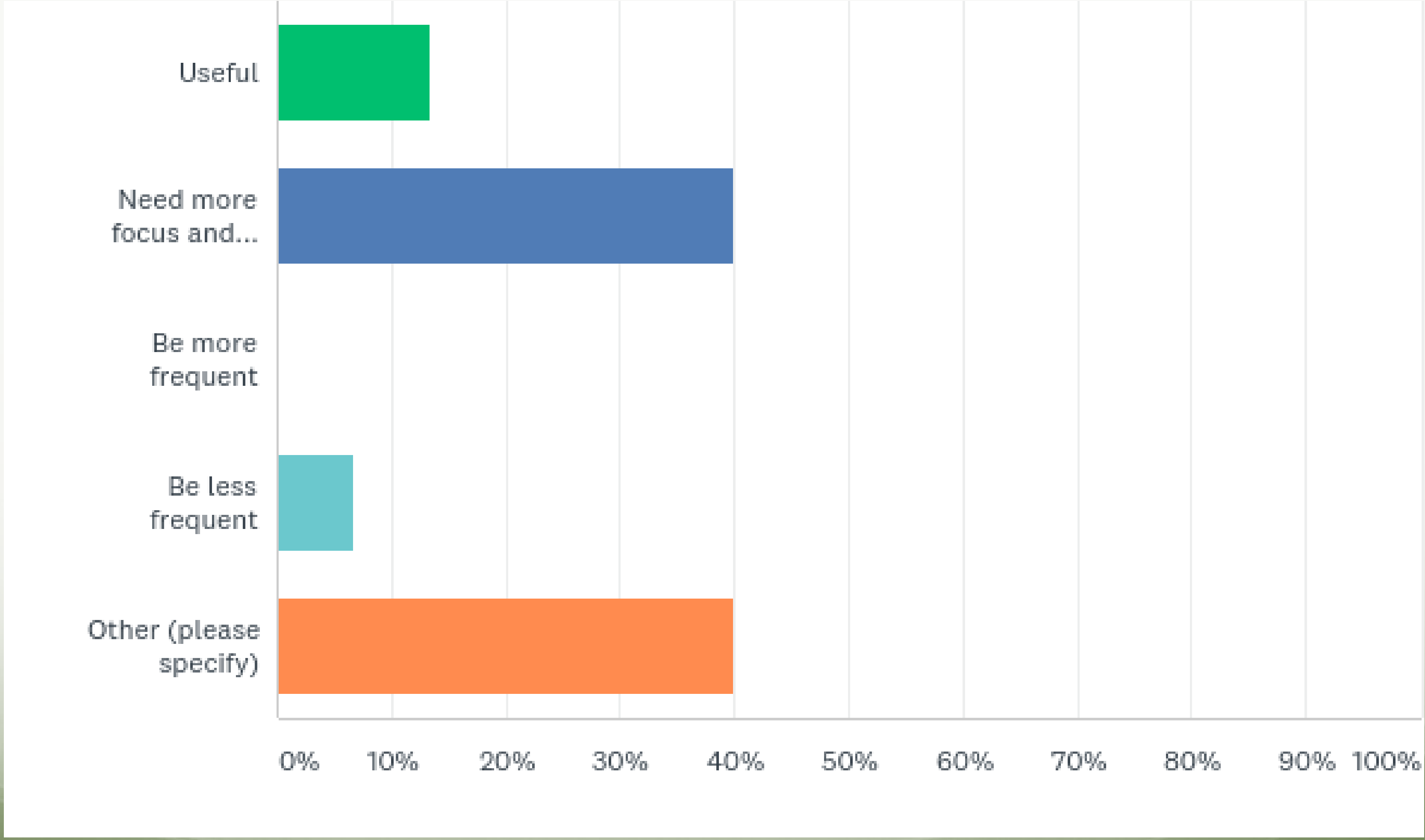
Q8: Confidence in NRC Advanced Reactors Licensing Schedule and Cost: What is your confidence that the NRC can transform its licensing process to provide greater schedule and cost certainty?



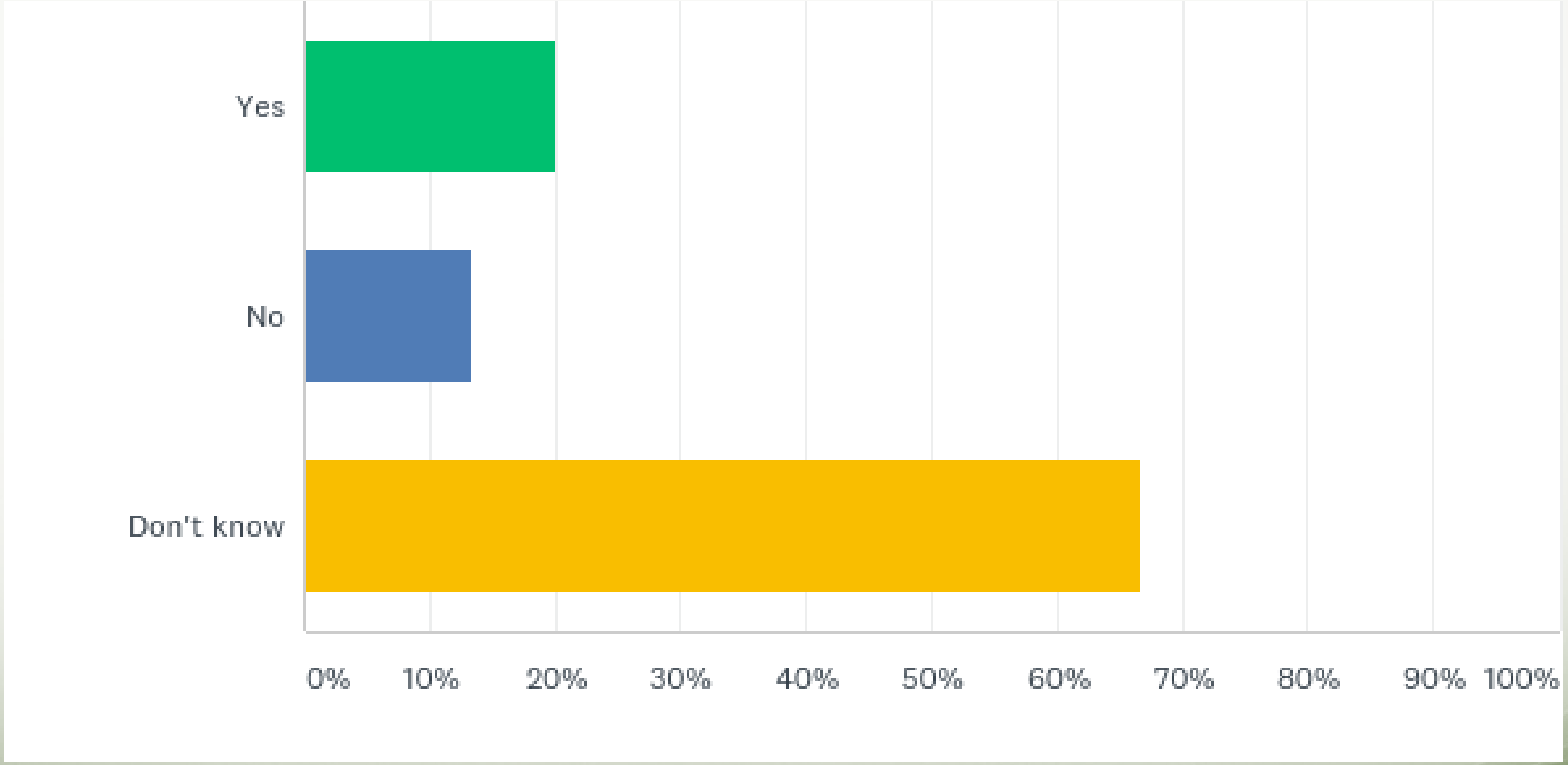
Q9: Should the NRC be doing more to seek non-fee based funding?



Q10: Value of NRC Advanced Reactor Stakeholder Meetings: Are the NRC's Stakeholder Meetings (held every 6-8 weeks)?

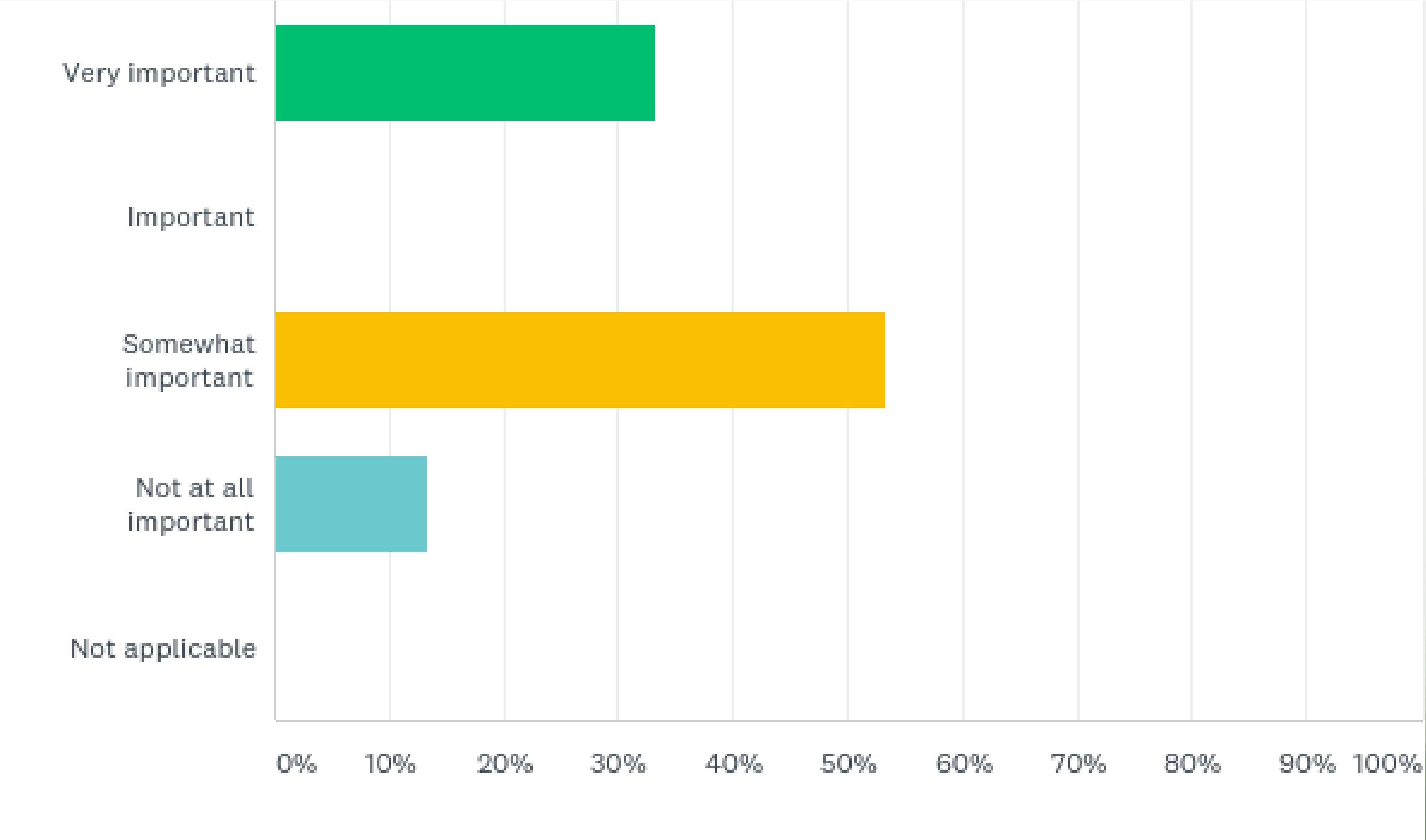


Q11: Do you believe the NRC Office of Research is putting sufficient time and resources towards Advanced Reactor development?



United States NIC Nuclear Industry Council

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

United States NIC Nuclear Industry Council

United States NIC Nuclear Industry Council

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

U.S. Nuclear Industry Council
1317 F Street, NW – Washington, DC 20004
(202) 332-8155 www.usnic.org

Meeting on Possible Regulatory Process Improvements for Advanced Reactors
December 13, 2018

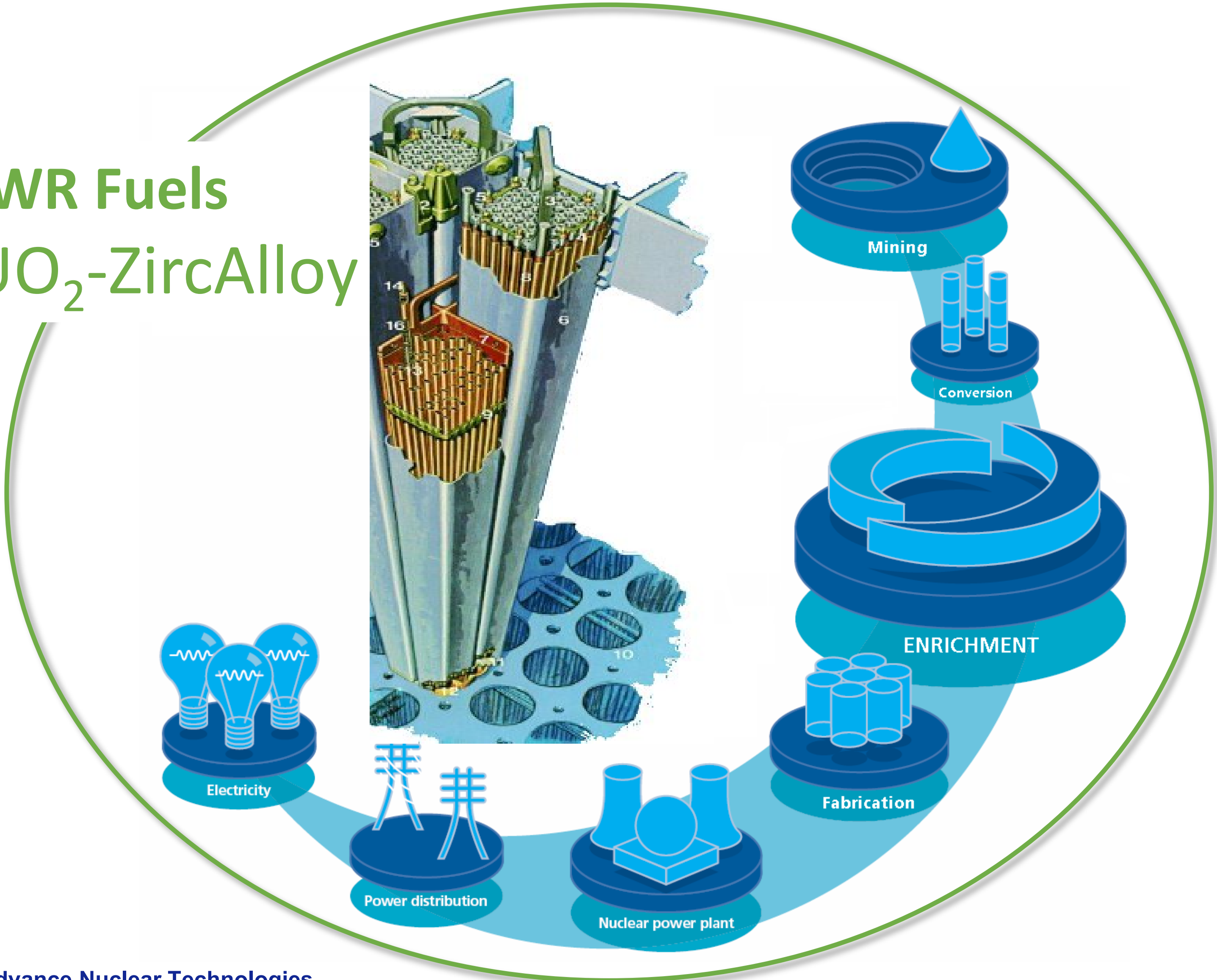
Next Generation Nuclear Fuels

Stephen Cowne, Chief Nuclear Officer, UUSA

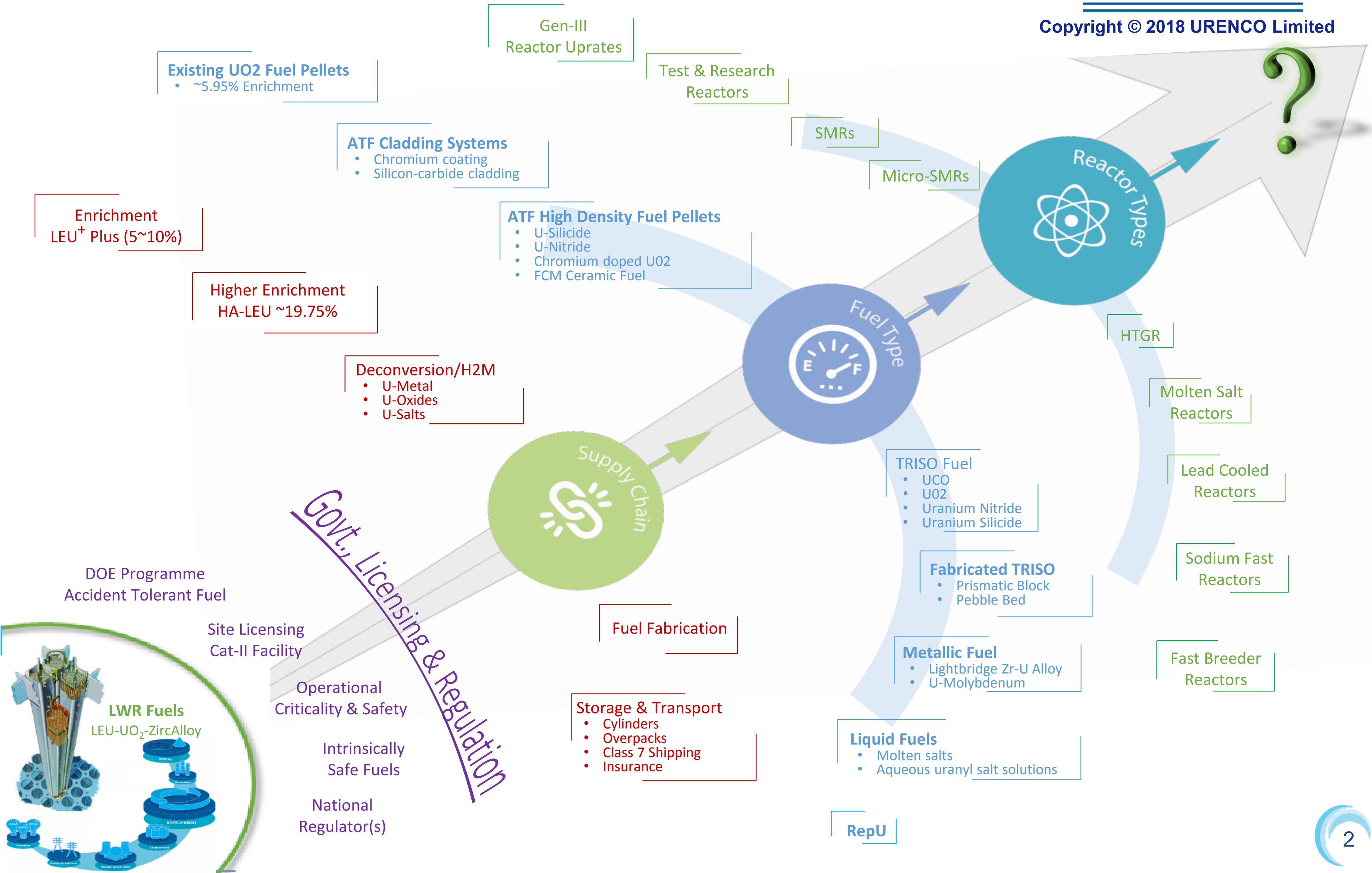
Today's Front-End Nuclear Fuel Cycle

LWR Fuels

LEU-UO_2-ZircAlloy



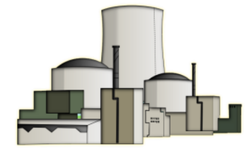
Next Generation Fuel Pathways: Range of options



The Future Nuclear Fuel Supply Chain

Existing Nuclear Fuel Supply Chain

Mining → U_3O_8 → Conversion → 0.711% UF_6 → Enrichment → <5% LEU → Fabrication → UO_2 → LWR Reactors UO_2 / ZircAlloy Fuels → Spent Fuel → Back End



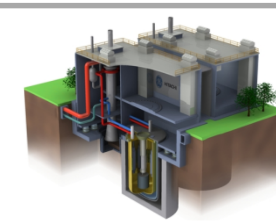
Completing the Future Nuclear Fuel Supply Chain

0.711% UF_6 → Enrichment → <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
UF_4 Salts
etc…

Gen III+, ATFs
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.

HA-LEU Fuel Cycle

- 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)

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

- 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.

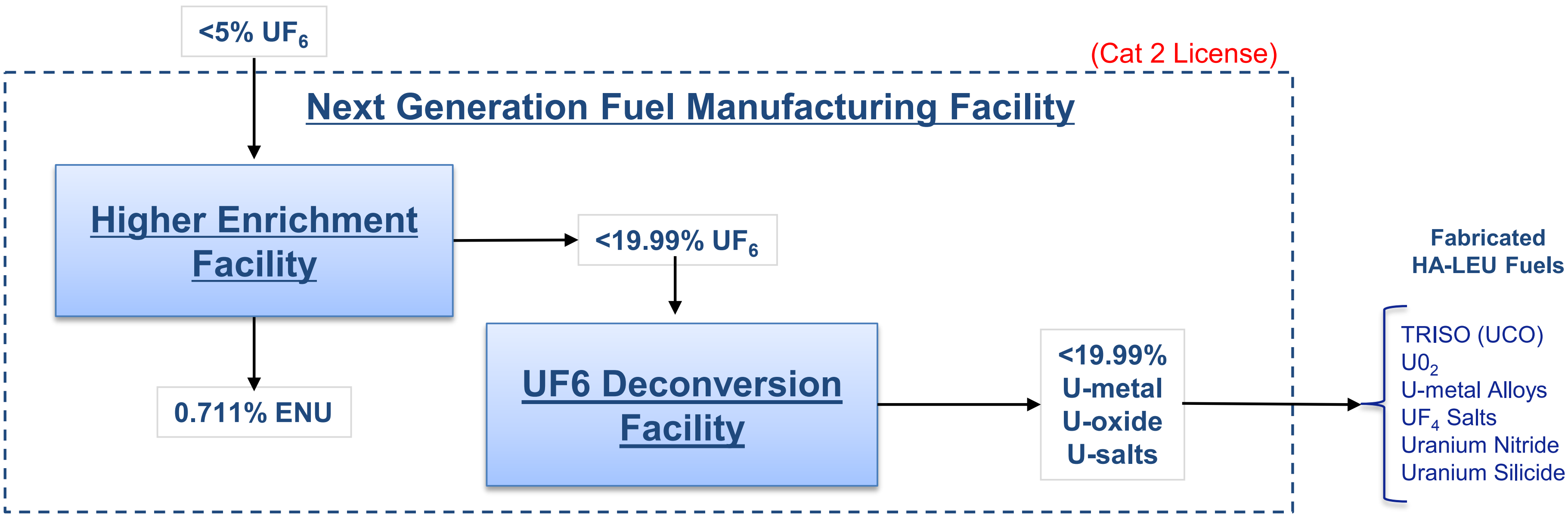
2-Box Model:
Co-location of Enrichment & Deconversion

Copyright © 2018 URENCO Limited

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 UF_6 above 5%

- Criticality benchmarking data is needed for HA-LEU assays
- For use with UO_2 fuel pellets

2b. UF_6 deconversion

- For other fuel types
- If existing transport packages are not approved at higher enrichments

HA-LEU Fuel Cycle: Licensing Challenges

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%.

HA-LEU Fuel Cycle: Initial Observations

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

Background

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

<u>NRC Advanced Reactor Policy Statement</u>

- 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

- SECY-11-0184, "Security Regulatory Framework for Certifying, Approving, and Licensing Small Modular Reactors."
  - 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."*

SECY-18-0076 Options

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.
  - 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.

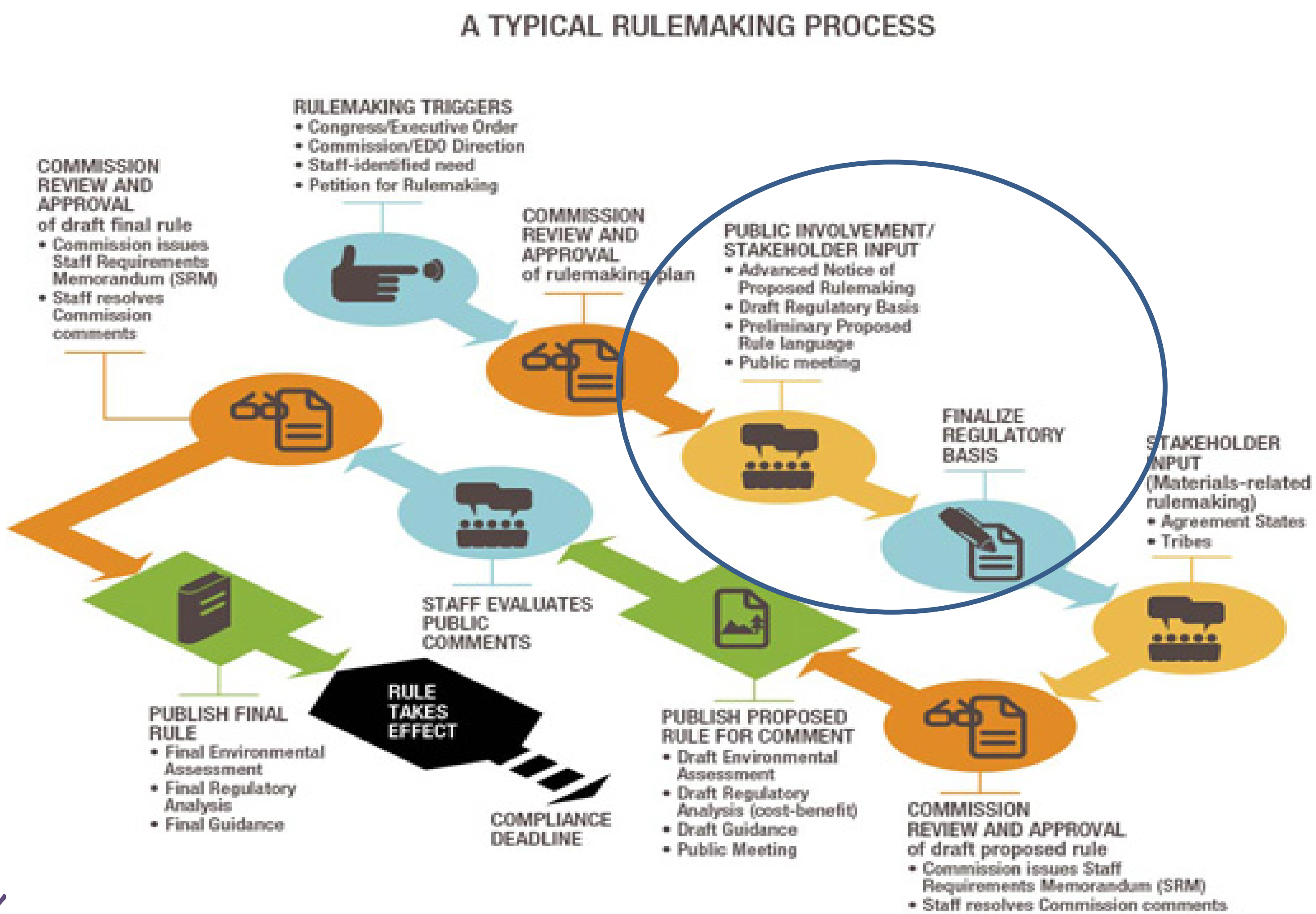
Staff Requirements Memorandum (SRM)

SRM Dated November 19, 2018

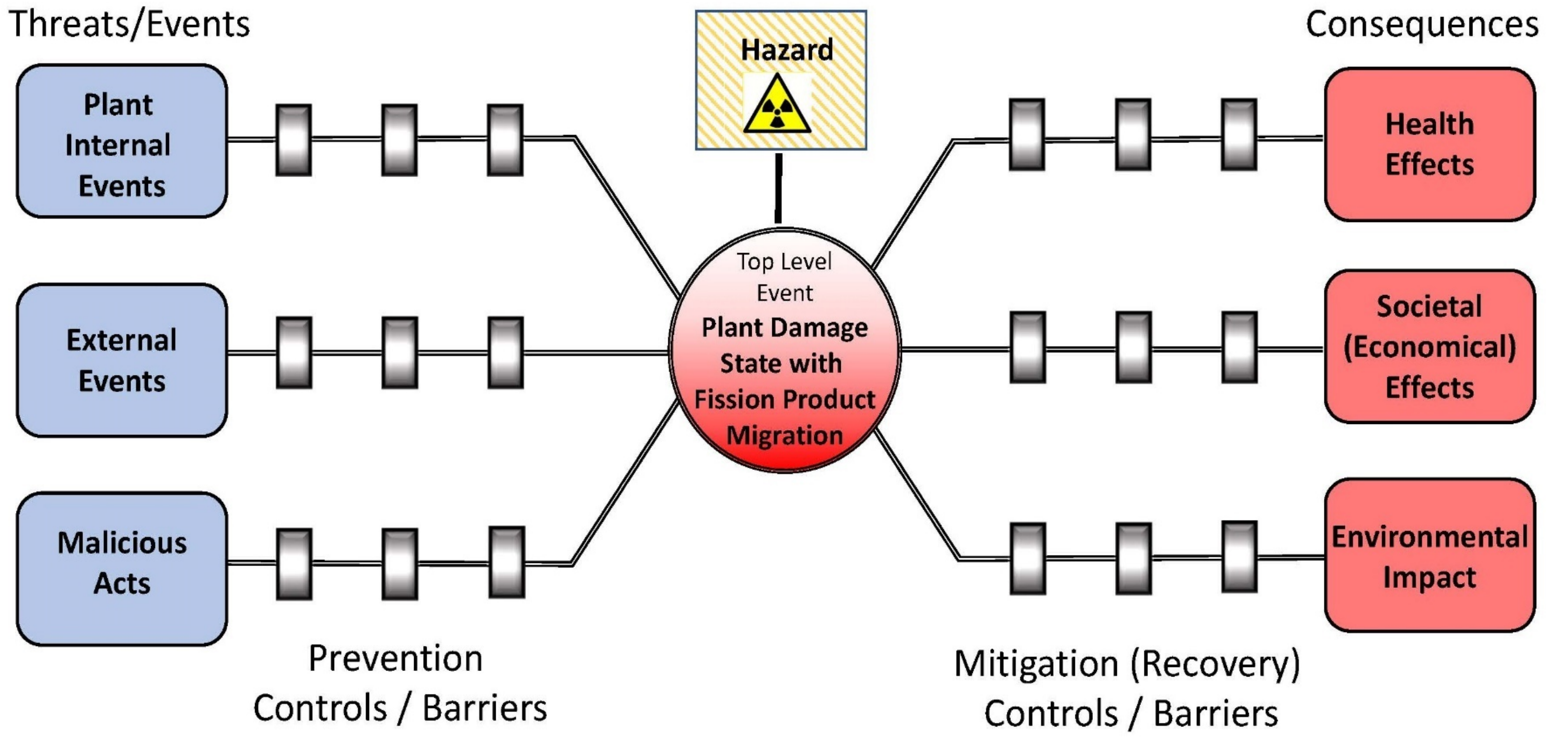
The Commission approved the staff's recommended Option 3, to initiate a limited-scope revision of regulations and guidance related to physical security for advanced reactors and approved the enclosed rulemaking plan, subject to the enclosed edits.

- 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



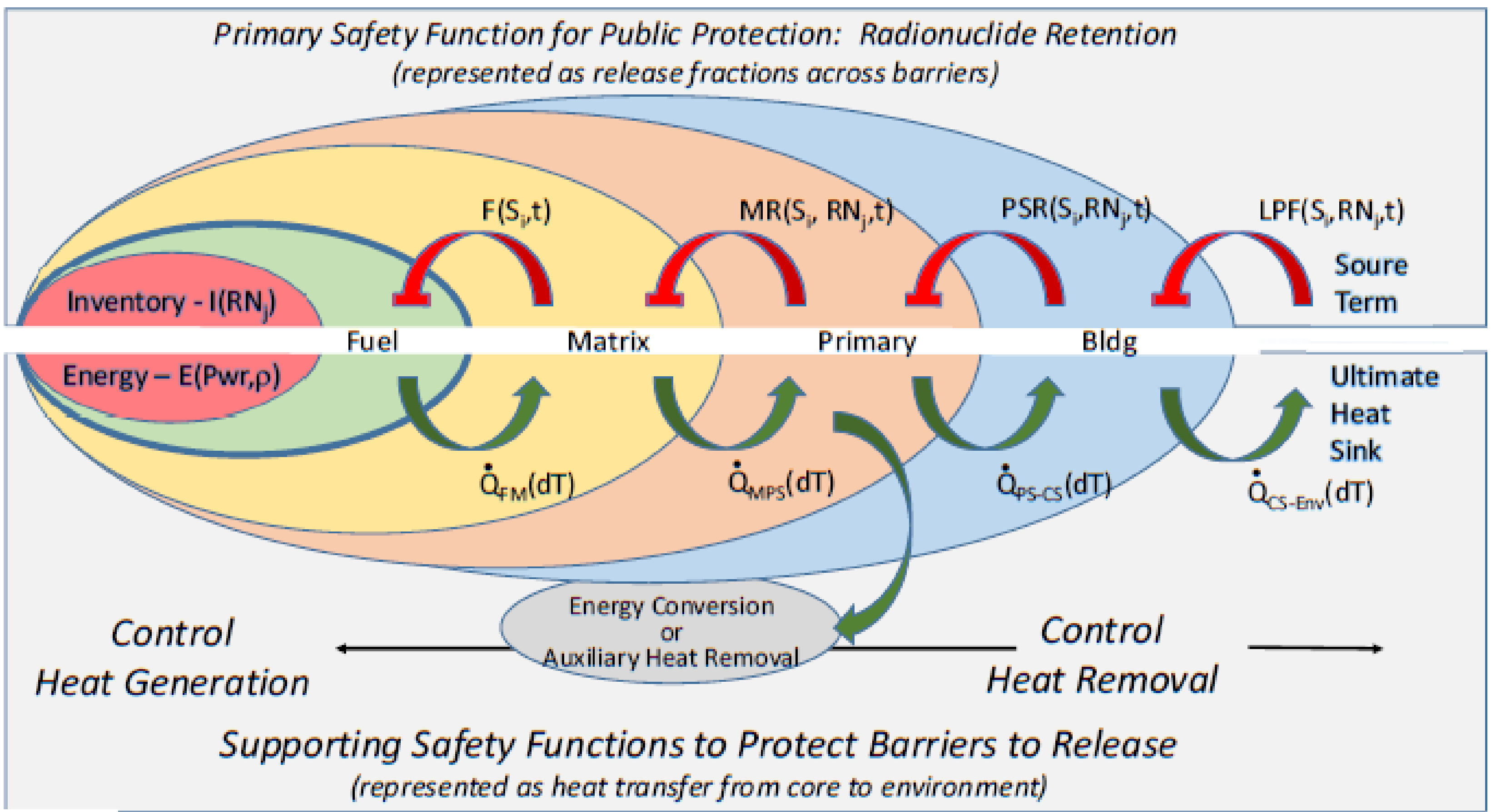
Barrier Assessment (Bow Tie Diagram)



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

Revisit First Principles

Fundamental Safety Functions and Mechanistic Source Term



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 ↓

(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 ↓

(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

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

Discussion

Potential Scope of Alternative Requirements

- 10 CFR 73.55(k) – armed responders
- 10 CFR 73.55(i) – secondary alarm stations
- ?
- ?
- ?

Advanced Reactor Program

Stakeholder Presentation/Discussion

NEI

Discussion

Stakeholder Presentation/Discussion
USUCS

General Discussion

Public Questions/Feedback