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United States Nuclear Regulatory Commission
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Public Meeting on Possible Regulatory Process Improvements for Advanced Reactor Designs

August 15, 2019

Advanced Reactor Program

Telephone Bridge: (800) 593-0695
Passcode: 3798624

Public Meeting

- Telephone Bridge

  800-593-0695

  Passcode: 3798624
- Opportunities for public comments and questions at designated times

Outline

9:00-9:10 am	Opening Remarks
9:10-9:40 am	Status Update
9:40-10:25 am	Technology Inclusive Content Applications Project (TICAP)
10:25-10:35 am	Break
10:35-11:00 am	Micro-Reactors Regulatory Topics
11:00-11:30 am	Concrete and Elevated Temperature
11:30-12:15 pm	Environmental Interim Staff Guidance and Generic EIS Developments
12:15-1:15 pm	Lunch
1:15-1:45 pm	10 CFR Part 53: Risk-informed, Technology Inclusive Regulatory Framework for Advanced Reactors Rulemaking
1:45-2:15 pm	Nuclear Energy Innovation and Modernization Act (NEIMA)
2:15-2:45 pm	Non-LWR Vision and Strategy Computer Code Reports: Modeling and Simulation of non-LWRs
2:45-3:15 pm	NRC Non-LWR Computer Code Development Plans
3:15-3:30 pm	Break
3:30-4:00 pm	Protecting Sensitive Information
4:00-4:30 pm	Open Discussion

- NRC Updates

- DOE Updates

- Technology Inclusive Content of Applications Project (TICAP)
  - Amir Afzali, Southern Company Services

Break

Meeting/Webinar will begin shortly

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- Micro-Reactors Regulatory Topics
  - Marc Nichol, NEI

- Concrete and Elevated Temperature
  - Madhumita Sircar, NRC

- Environmental Interim Staff Guidance and Generic EIS Developments
  - Donald Palmrose/Mallecia Sutton, NRC

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Lunch

Meeting/Webinar will begin shortly

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- [10 CFR Part 53: Risk-Informed, Technology Inclusive Regulatory Framework for Advanced Reactors Rulemaking]

  \- William Reckley, NRC

Advanced Reactor Program

- Nuclear Energy Innovation and Modernization Act (NEIMA)
  - NRC Section 103 Activities
    - John Segala, NRC
  - Establishing Metrics and Milestones
    - Steven Lynch, NRC

- Non-LWR Vision and Strategy Computer Code Reports
  - Stephen Bajorek, NRC
- NRC Non-LWR Computer Code Development plans for severe accident progression, source term, and consequence analysis
  - Hossein Esmaili, NRC

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Break

Meeting/Webinar will begin shortly

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Advanced Reactor Program

- Protecting Sensitive Information
  - Stewart Magruder, NRC

2019 Tentative Schedule for Periodic Stakeholder Meetings
October 10
December 12

Open Discussion and Closing

TICAP

Technology Inclusive Content of Application Project

Amir Afzali
Southern Company

NRC Advanced Reactor Public Meeting Briefing
August 15, 2019

Introduction

- Project Purpose: To collaboratively work with the ARRTF and NRC to develop a technology inclusive formulation for preparing content of application that will have the following attributes:
  - Versatile - The variance in technologies and designs requires robust application guidance that is versatile enough to be used by most if not all potential applicants
  - Systematic - It facilitates thorough and consistent safety assessments for different designs across and within different technologies
  - Compatible - It correlates to the underlying safety basis/objective of the current light-water centric content of application, thereby demonstrating consistency with the Commission's mission of protecting public safety

Status and Important Inputs

- Status
  - Team formed
  - Kickoff meeting July 18, 2019
  - Second meeting August 8, 2019
  - Focus on developing Project Plan (due September 30)
  - Plan to Propose a set of fundamental safety functions by 12/20/2019
    - Product or process/method for NRC's interaction (e.g., WP for NRC's official review) not yet defined.
- Necessary Event
  - Regulatory Guide endorsing NEI 18-04 – Expected Date Dec. 2019

TICAP Success Criteria and Guiding Principles

Success Criteria

Provide a document that outlines the content of an application in a manner that is technology inclusive, uses LMP methodology and can be submitted to NRC for endorsement within two calendar years from initiation of Phase 2.

Guiding Principles

- Right-size content of application to be commensurate with the complexity of the applicant’s safety case.
- Improve the overall efficiency of review and optimize generation of application’s information in terms of scope and level of details.
- Is limited to utilizing the outputs developed through application of the Licensing Modernization Project process defined in NEI 18-04
- Will be technology inclusive
- **Will include a formulation that facilitates meeting the underlying safety basis/objective of the current regulations** (provide an alternative to show compliance with the underlying safety basis/objective vs. creating equivalency) -
    - For certain designs this mean certain information required as part of application for a LWR may not be included
    - **May result in regulatory questions being raised (expect conversations around 2020, 3rd quarter).**
- Should contribute beneficially to the development of the NRC’s new regulatory framework for advanced reactors (10 CFR Part 53)

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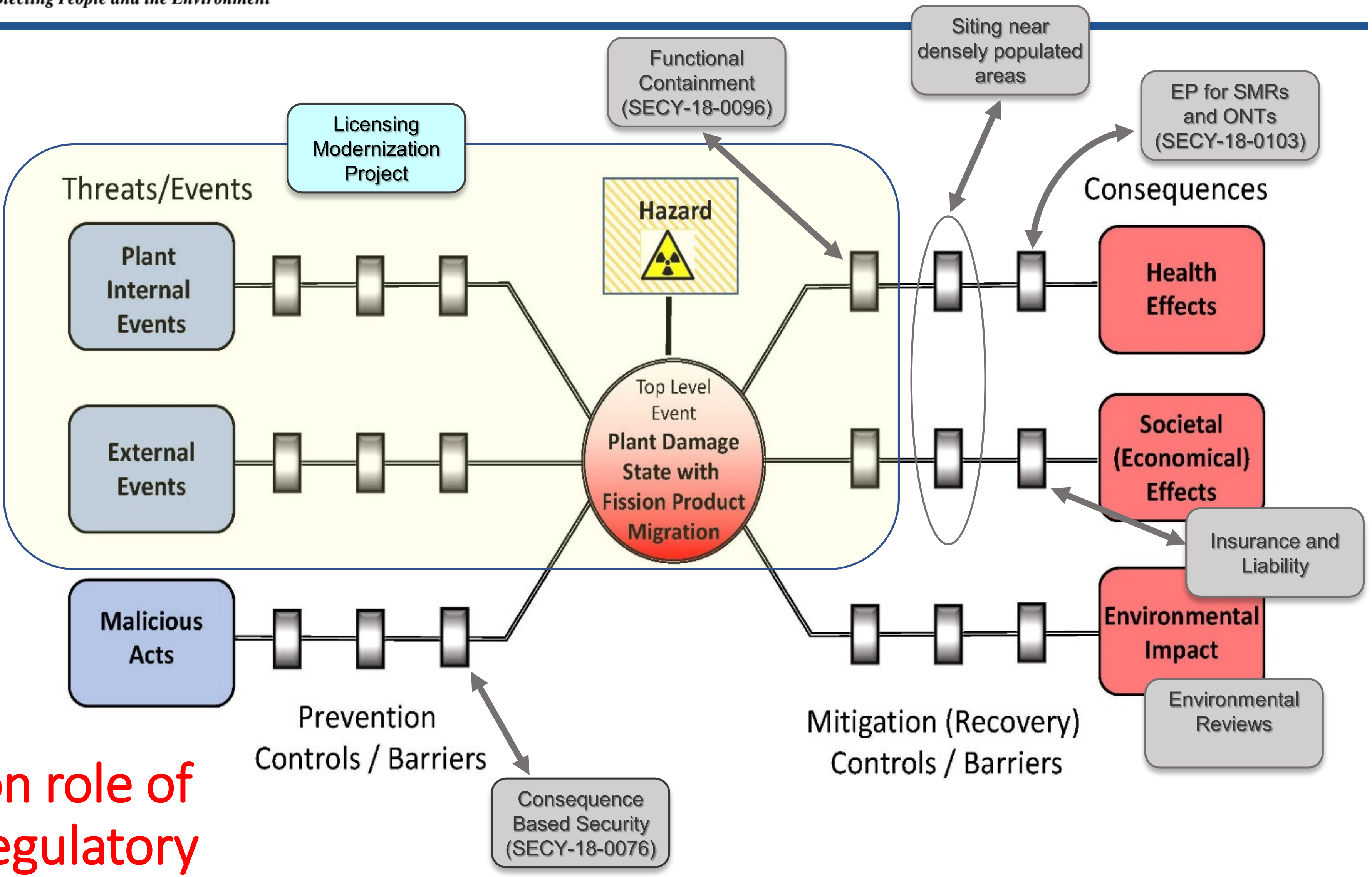
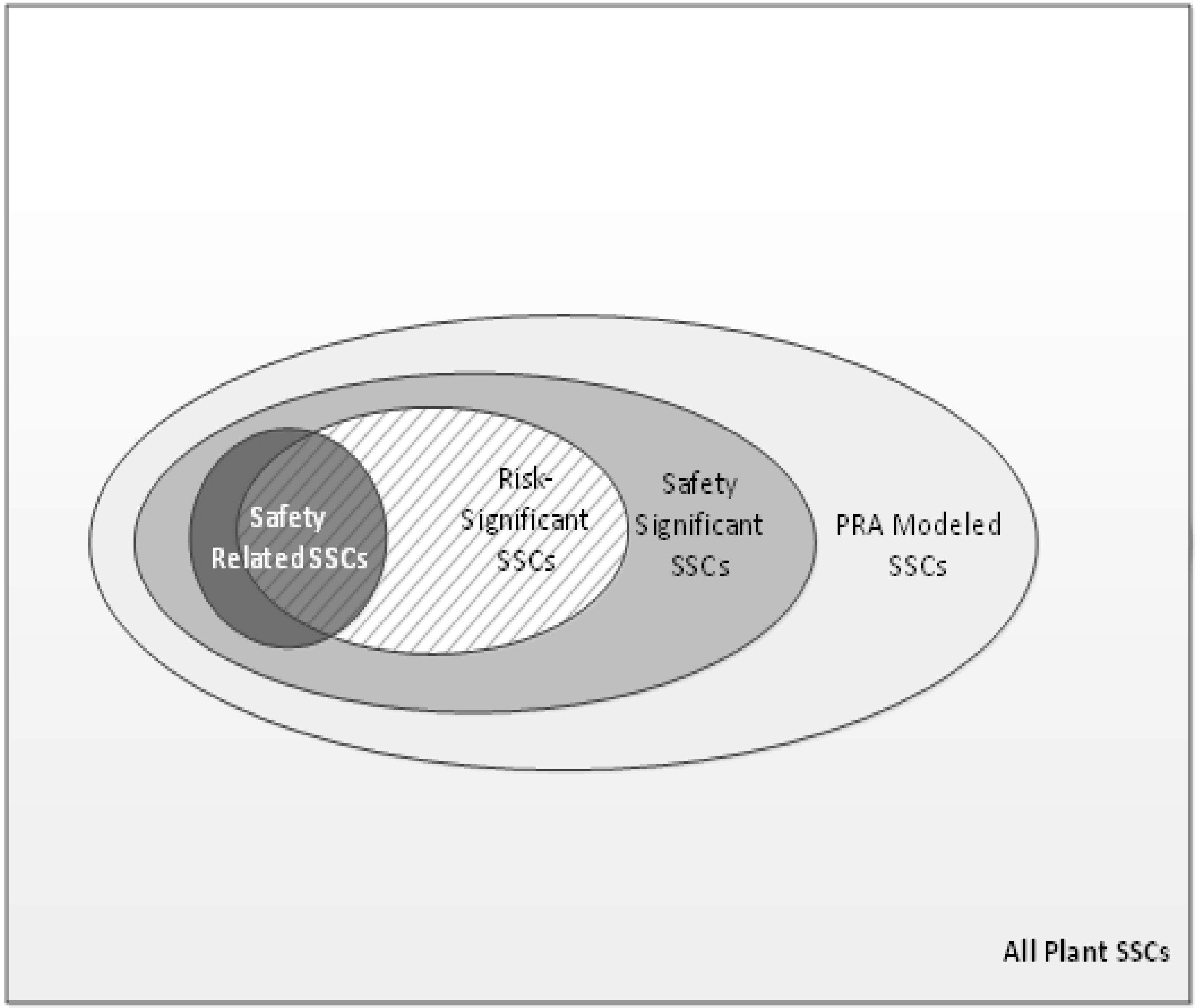


Simplified representation role of LMP within regulatory framework

Fundamental Consideration Based on Use of LMP Process

TICAP will propose what level of information is needed for SSCs and their associated programs for each class of SSC



?

Action Plan for Reviewing and Endorsing Non-LWR PRA Standard

Advanced Reactor Stakeholder Meeting

August 15, 2019

Outline of Plan

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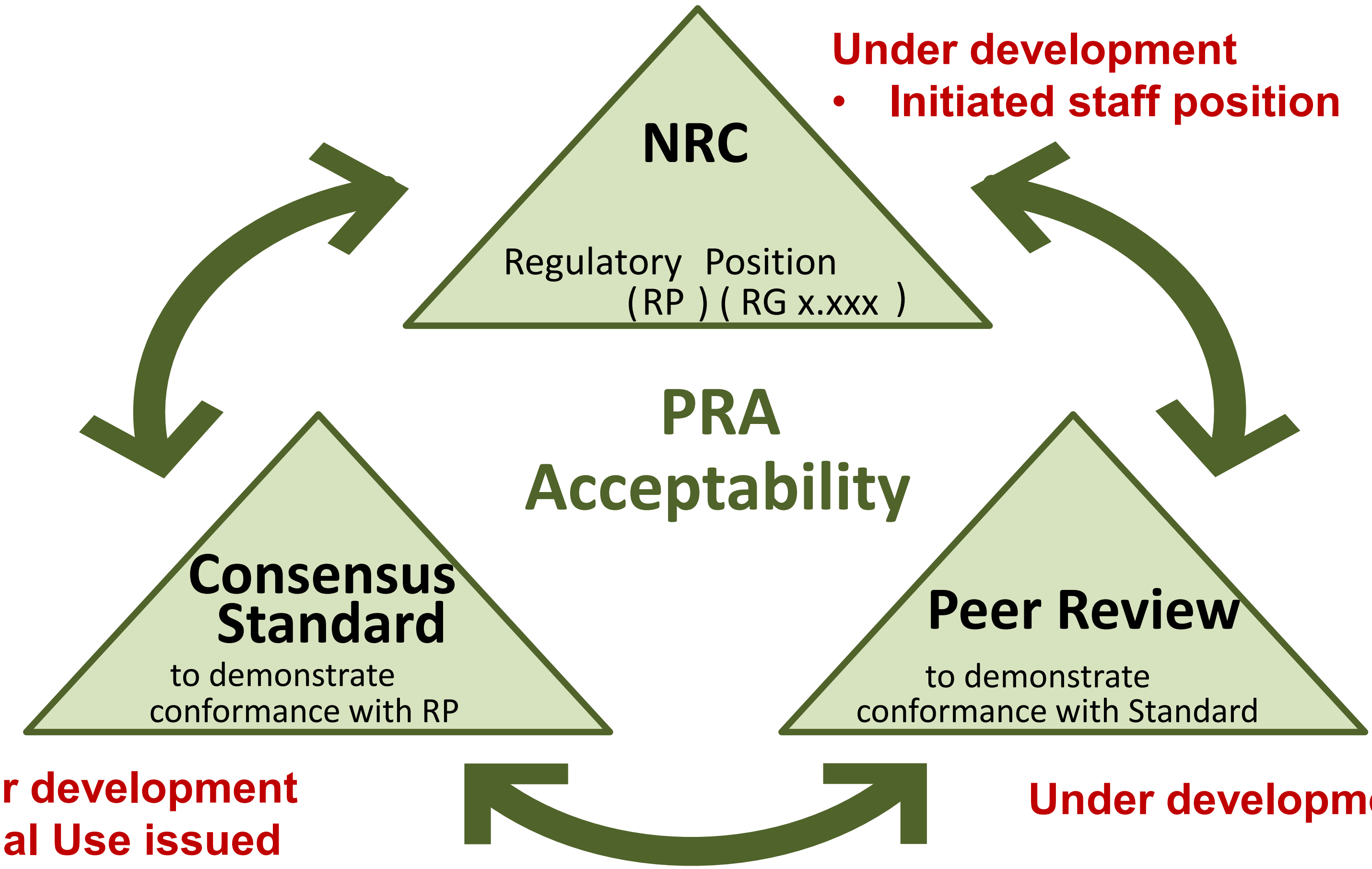
- Objectives and Scope
- Tasks
  - Task 1: Supporting development of the standard
  - **Task 2: Preparation for review of the standard**
  - **Task 3: Reviewing the standard**
  - Task 4: Maintaining PRA standard
  - **Task 5: Development of schedule**
  - Task 6: Identification of resources
  - Task 7: Development of communication plan

Task 2: Preparation for Review of the Standard – Involves 6 Subtasks

- Subtask 2-4: Develop staff position for an acceptable non-LWR PRA (task initiated)
  - Define the objectives for each technical element, considering the different applications
  - Define the technical attributes and characteristics needed to accomplish the objective
  - Develop the staff position on an acceptable peer review process, addressing an acceptable peer review process, team qualifications, and documentation
- Subtask 2-5: Identification and resolution of technical and policy issues
  - Review each technical element associated with each risk level and hazard, for each application type, and identify possible technical or policy issues
  - Describe the significance of the issues
  - Identify whether there is ongoing research to address the issues and what research are needed
- Subtask 2-6: Guidance for staff review of non-LWR PRA standard for endorsement (draft completed)
  - Guidance on how to approach the review
  - Criteria for determining acceptance (no objection)

Status of NRC RG, ASME/ANS Standard, and NEI Guidance



Under development
- **Initiated staff position**

Under development
- **Trial Use issued**
- **Final, publication in 2020/2021**

Under development ?????

Micro-Reactors

Regulatory Topics

Marc Nichol, Director New Reactor Deployment

August 15, 2019

NUCLEAR ENERGY INSTITUTE

Micro-Reactor Regulatory Issues

Priority Issues	Addressed in Broader Efforts	Non-Urgent
1. Review Scope, Duration, Level of Effort 2. Aircraft Impact 3. Operations (auto/remote) 4. Resident Inspector 5. Physical Security 6. Emergency Preparedness	• Siting • Environmental Reviews	• Transportation • Annual Licensee Fees • Fuel • Generic License • PRA • QA

No issues identified to-date

- Liability Insurance
- Decommissioning Funding

Unique Micro-Reactor Considerations*

- Typically 1 MWe to 10 Mwe
- Very small size
    - Site <0.1 acres, building ~size of a house, reactor fits in shipping container
- Very small potential consequences
    - Source terms as low as 1% of today's reactors
    - Fail-safe: shuts itself off, cannot meltdown
    - Proliferation resistant fuel
- Operational simplicity
    - Few to zero moving parts
    - Automatic operations
    - Minimal maintenance

*General description, all features may not be applicable to all designs

Environmental Interim Staff Guidance and Generic EIS Developments

Donald Palmrose, PhD
Sr. Reactor Engineer

Mallecia Sutton
Sr. Project Manager

Office of New Reactors
U.S. Nuclear Regulatory Commission

Advanced Reactor Stakeholders Meeting
August 15, 2019

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Agenda

- Staff seeking input on:
  - Draft ISG
  - GEIS
  - Possible EA for some Advanced Reactors
- Open discussion

Interim Staff Guidance

Environmental Considerations Associated with Micro-reactors

- Provides supplemental staff guidance for the environmental review to address differences with large LWRs:
  - Smaller footprint affects fewer resources
  - May not use cooling water
  - Smaller rad and non-rad waste streams
  - Reduced socioeconomic impacts
  - Smaller size generally translates to fewer impacts

Interim Staff Guidance (cont.)

- Provides guidance for how to “incorporate by reference,” or IBR, for environmental reviews
- Reducing duplication of effort, size of documents while maintaining quality to meet NRC’s NEPA obligations
- Issuance of a draft ISG for comment and use by December 2019

Considerations for Advanced Reactor Reviews

- GEIS feedback
  - Review of previous GEISs for benefits, costs, and limitations
  - Appropriate scope acceptance criteria
  - Enough publicly available data
  - Staff resource assessment
  - Need for new supporting studies
  - Need for rulemaking
  - What would be a realistic schedule

Considerations for Advanced Reactor Reviews

- EIS versus EA feedback
- Current practice under 10 CFR 51.20
- Can an EA address some advanced reactor reviews
  - 10 CFR 51.21
  - Rulemaking needed?
  - Considering staff guidance

Open Discussion

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Concrete and Elevated Temperature

Madhumita Sircar*, Jose Pires*, Ata Istar**
U.S. Nuclear Regulatory Commission

*Office of Nuclear Regulatory Research
**Office of New Reactors

Advanced Reactors Stakeholders Meeting
Nuclear Energy Institute
Washington, DC.
August 15, 2019

Scope of Presentation:

- Effects of High Temperature on Concrete

Current Code Requirements by ACI 349:

- Provision E.4, “Concrete Temperature,”

  150^{o}F for Normal Operations

  350^{o}F for Accident or Short-Term Period

Key References

- **NUREG/CR-6900**, *"The Effect of Elevated Temperature on Concrete Materials and Structures - A Literature Review. US NRC, March 2006."*

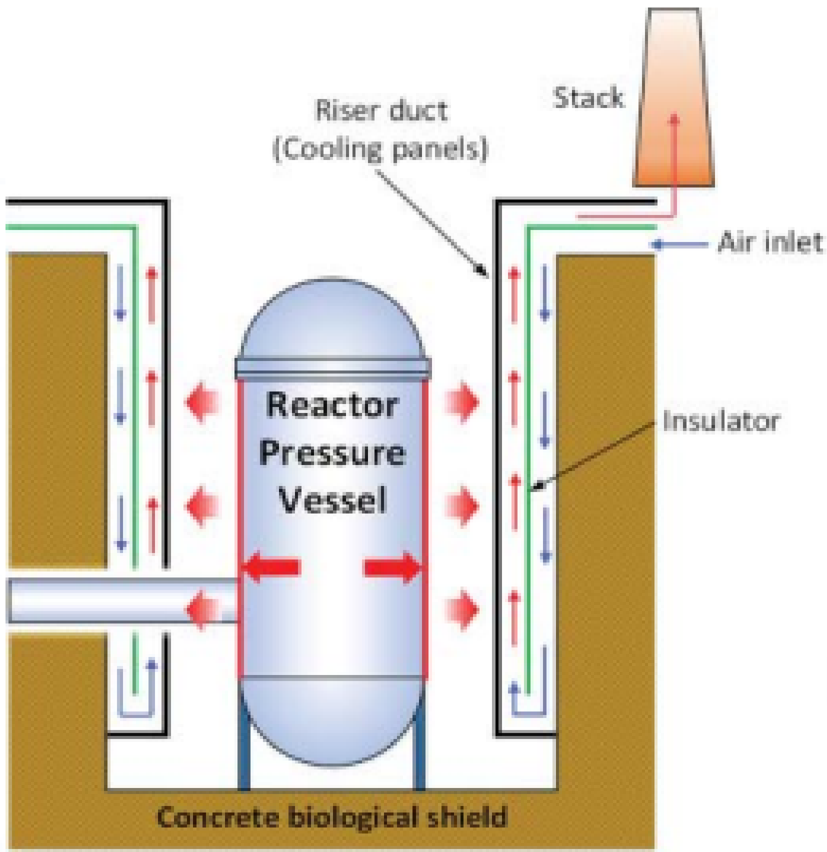
- **NUREG/CR-7031**, *"A Compilation of Elevated Temperature Concrete Material Property Data and Information for Use in Assessments of Nuclear Power Plant Reinforced Concrete Structures. US NRC, December 2010."*

Advanced Reactor Concrete Structures under High Temperature

- Concrete Reactor Building which may be one of the Functional Containments
  - Independent barrier provide defense-in-depth

- Other Concrete Structures

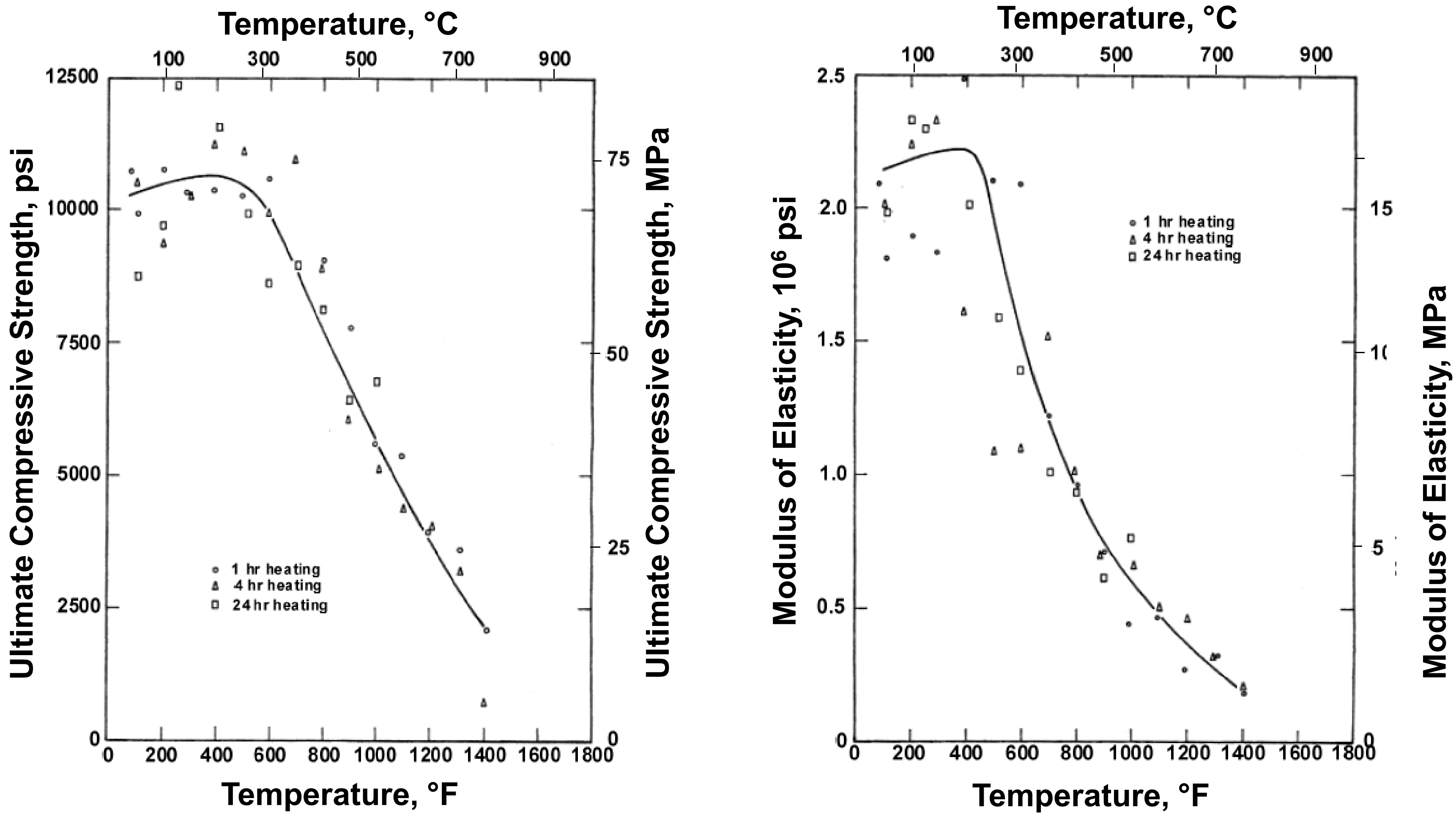
Example of a Passive Cooling System for Concrete Structures under High Temperature



Independent barriers provide defense-in-depth

[Source: NRC HTGR training prepared by INL]

Concrete under High Temperature



Ultimate compressive strength and modulus of elasticity of Type I Portland cement paste (w/c:0.33)

[Source: Harmathy et al. Fig. 1 and 2 as referenced in NUREG/CR-6900]

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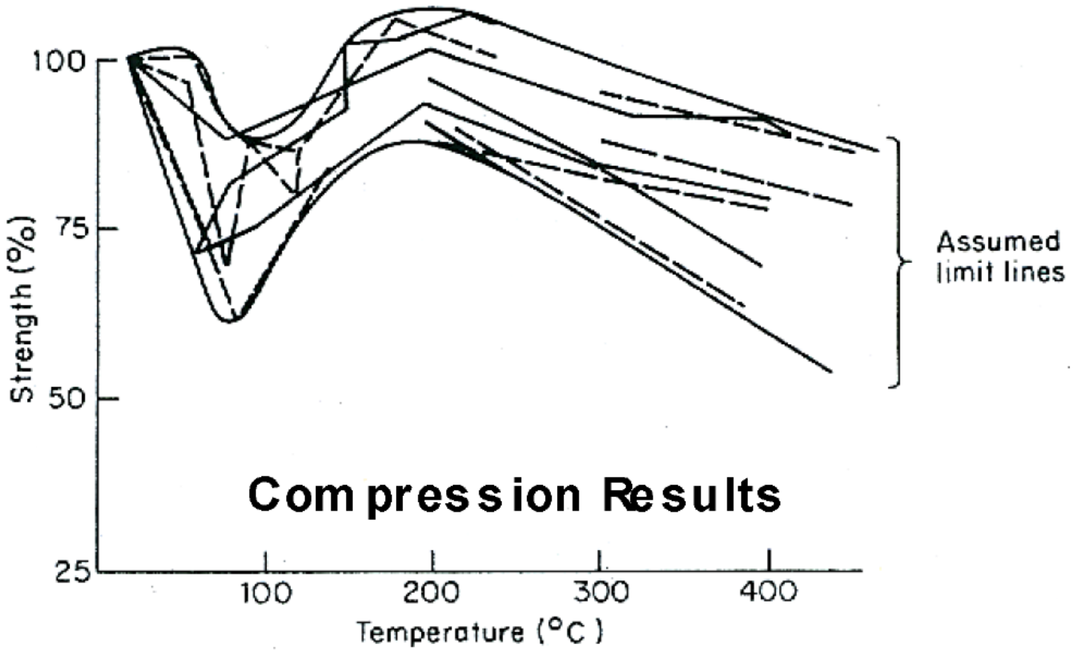
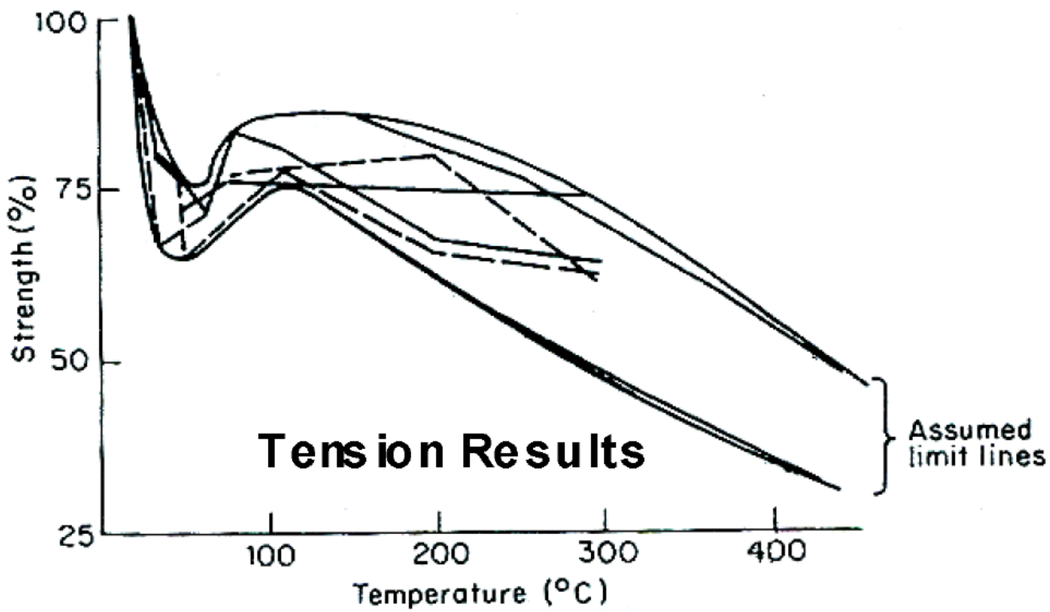
Concrete under High Temperature, cont.



Comparison of Effect of Elevated Temperature on the Compressive Strength of Concretes Fabricated using Different Types of Conventional Aggregate Materials

[Source: Blundell et al. Fig. 2.121 as referenced in NUREG/CR-7031]

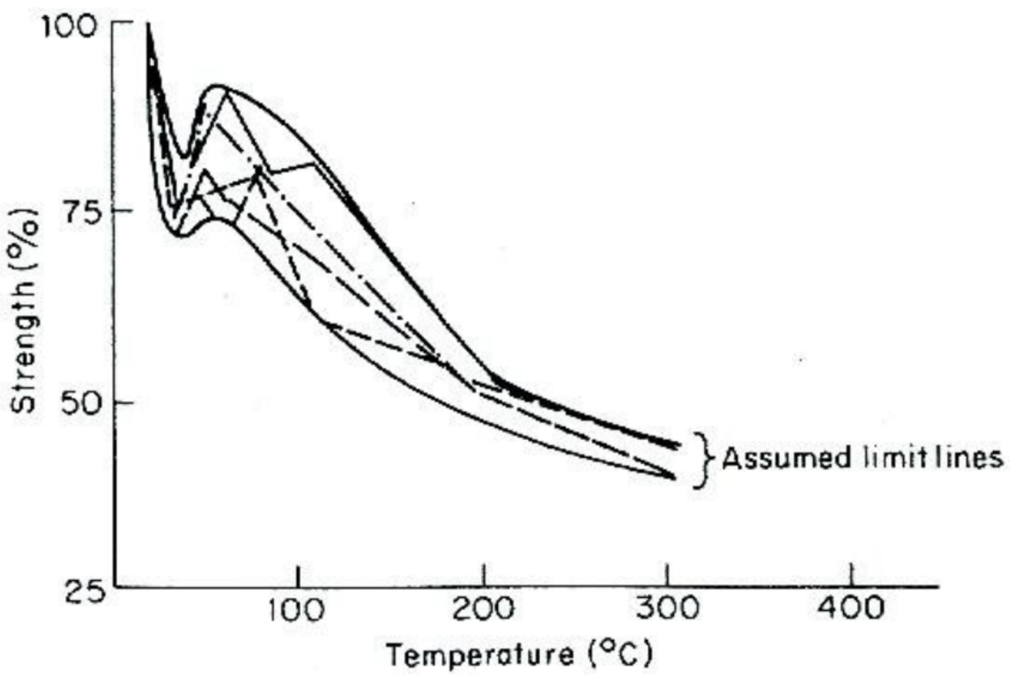
Concrete under High Temperature, cont.



Comparison of the Effect of Elevated Temperature on the Tensile Strength of Concretes Fabricated using Different Types of Conventional Aggregate Materials

[Source: Blundell et al. Fig. 2.121 as referenced in NUREG/CR-7031]

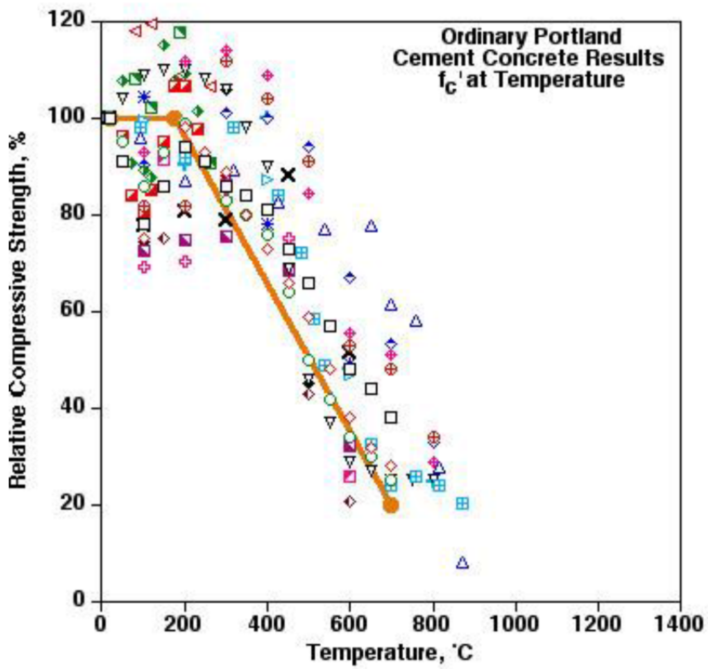
Concrete Under High Temperature, cont.



Comparison of Effect of Elevated Temperature on the Relative Bond Strengths of Mild Steel to Concretes Fabricated using Different Types of Conventional Aggregate Materials

[Source: Sullivan Fig. 2.174 as referenced in NUREG/CR-7031]

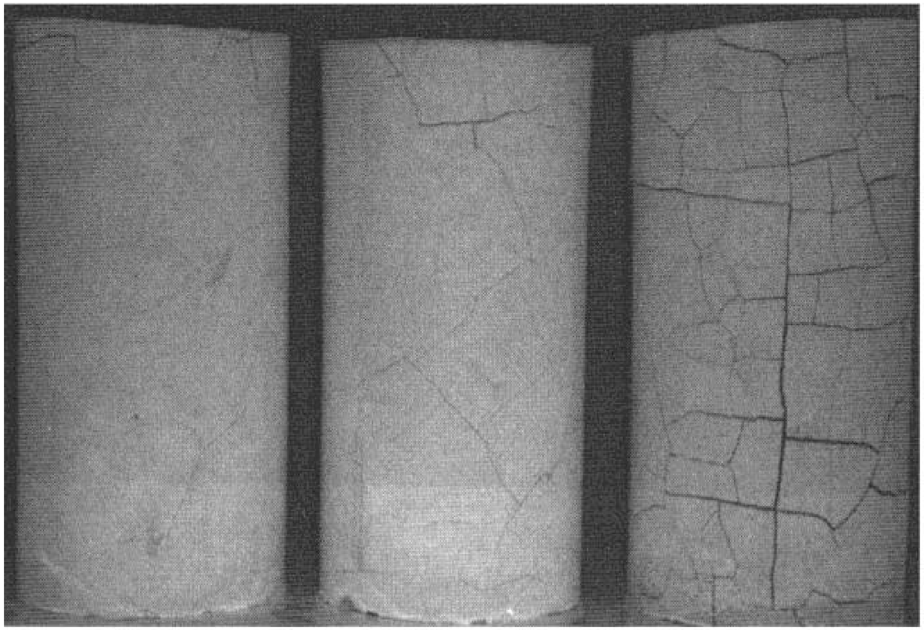
Concrete under High Temperature, cont.



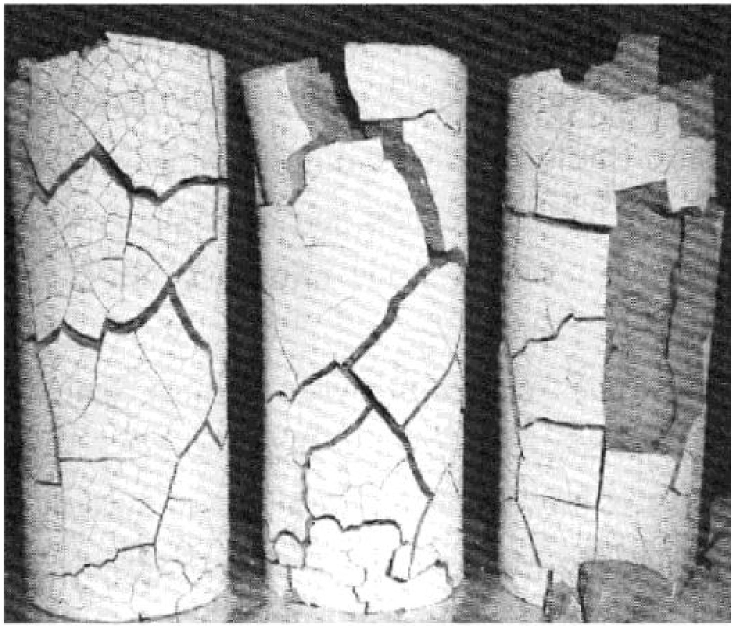
The Effect of Temperature on the Compressive Strength of Portland Cement Concrete

[Source: Fig. 2.86 as referenced in NUREG/CR-7031]

Concrete under High Temperature, cont.



Ordinary Portland Cement Paste Specimens Immediately After 800°C



Ordinary Portland Cement Paste Specimens 2 Days After 800°C

[Source: Fig. 2.6 as referenced in NUREG/CR-7031]

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Summary

- Prevention of elevated temperature in concrete
- Preferably meeting ACI 349 code limits
- Reliability of preventive systems
- Condition monitoring
- Other considerations
  - Thermal cycling
  - Normal and accident conditions

THANKS

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10 CFR Part 53: Risk-informed, Technology Inclusive Regulatory Framework for Advanced Reactors Rulemaking

Advanced Reactor Stakeholder Meeting

August 15, 2019

Nuclear Energy Innovation and Modernization Act (NEIMA)

- NEIMA Section 103 requires that the NRC "complete a rulemaking to establish a technology-inclusive, regulatory framework for optional use by commercial advanced nuclear reactor applicants for new reactor license applications."
- Rulemaking is to be completed no later than December 31, 2027.

NEIMA

- NEIMA defines “advanced nuclear reactor” as “a nuclear fission or fusion reactor, including a prototype plant . . . with significant improvements compared to commercial nuclear reactors under construction” as of January 14, 2019, including improvements such as additional inherent safety features; significantly lower levelized cost of electricity; lower waste yields; greater fuel utilization; enhanced reliability; increased proliferation resistance; increased thermal efficiency; or ability to integrate into electric and nonelectric applications.

NRC Staff Activities

- NRC has formed a staff working group
- Working group is drafting a rulemaking plan and formulating initial thoughts on scope of rule
- Today's meeting marks first staff outreach to external stakeholders
- A focused public workshop is being planned for October 2019

Rule Applicability

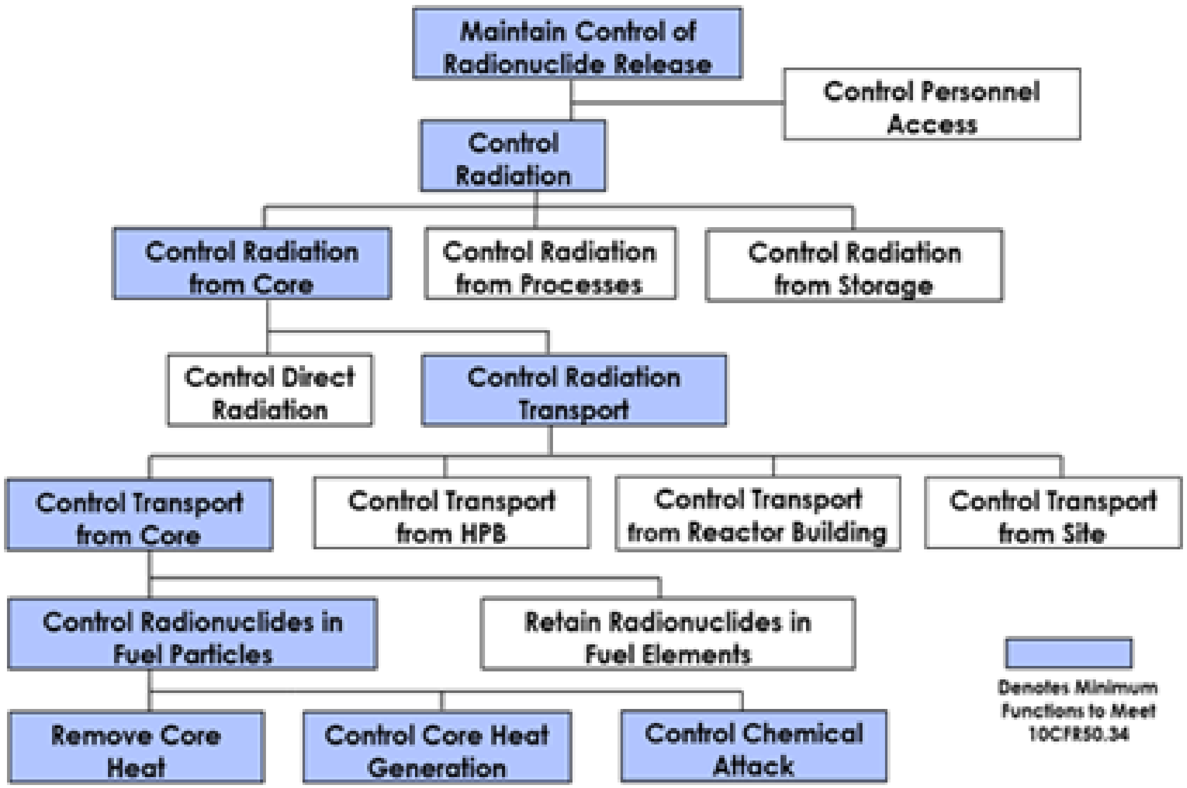
- NEIMA's definition of "advanced nuclear reactor" covers:
  - Light-water small modular reactors
  - Non-light-water reactors (non-LWRs)
  - Fusion reactors
- The staff interprets NEIMA as not requiring the rulemaking to cover reactor technologies similar to current operating reactors or Generation III+ large LWRs

Questions for Discussion

- Question: Which types of requirements should be included?
  - Technical requirements equivalent to 10 CFR Part 50?
  - Licensing processes equivalent to 10 CFR Parts 50 & 52?
  - Complete plant/license life cycle or initial license applications?
  - Level of detail for technical requirements
  - All technical requirements, including security and emergency preparedness?

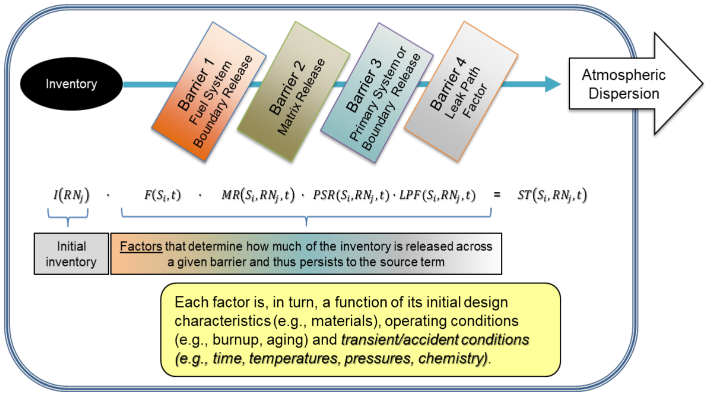
Developing Functional Requirements

Next Generation Nuclear Plant (NGNP) Concepts



Licensing Modernization Project

Recent NRC activities related to advanced reactors (e.g., functional containment performance criteria, possible changes to emergency planning & security, and DG-1353) recognize the limitations of existing LWR-related guidance, which requires a return to first principles such as fundamental safety functions supporting the retention of radionuclides



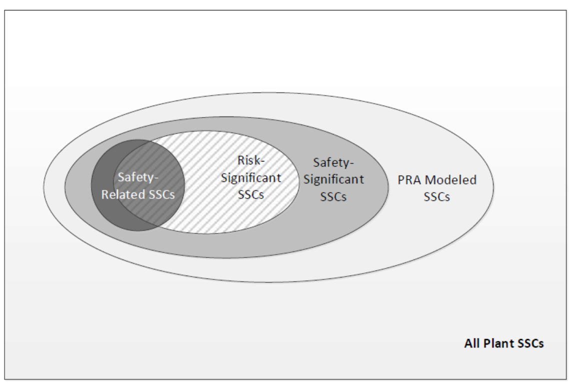
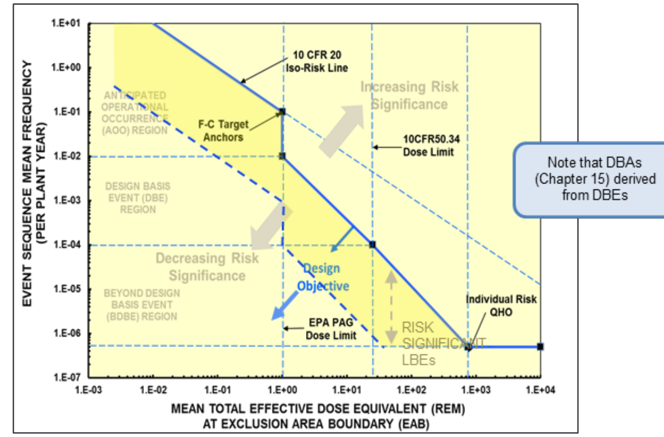
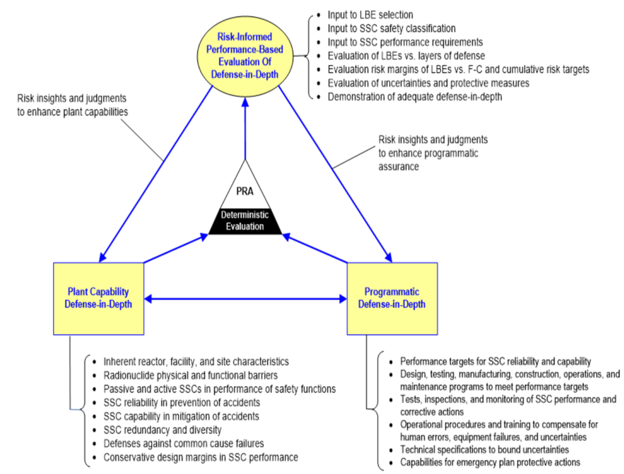


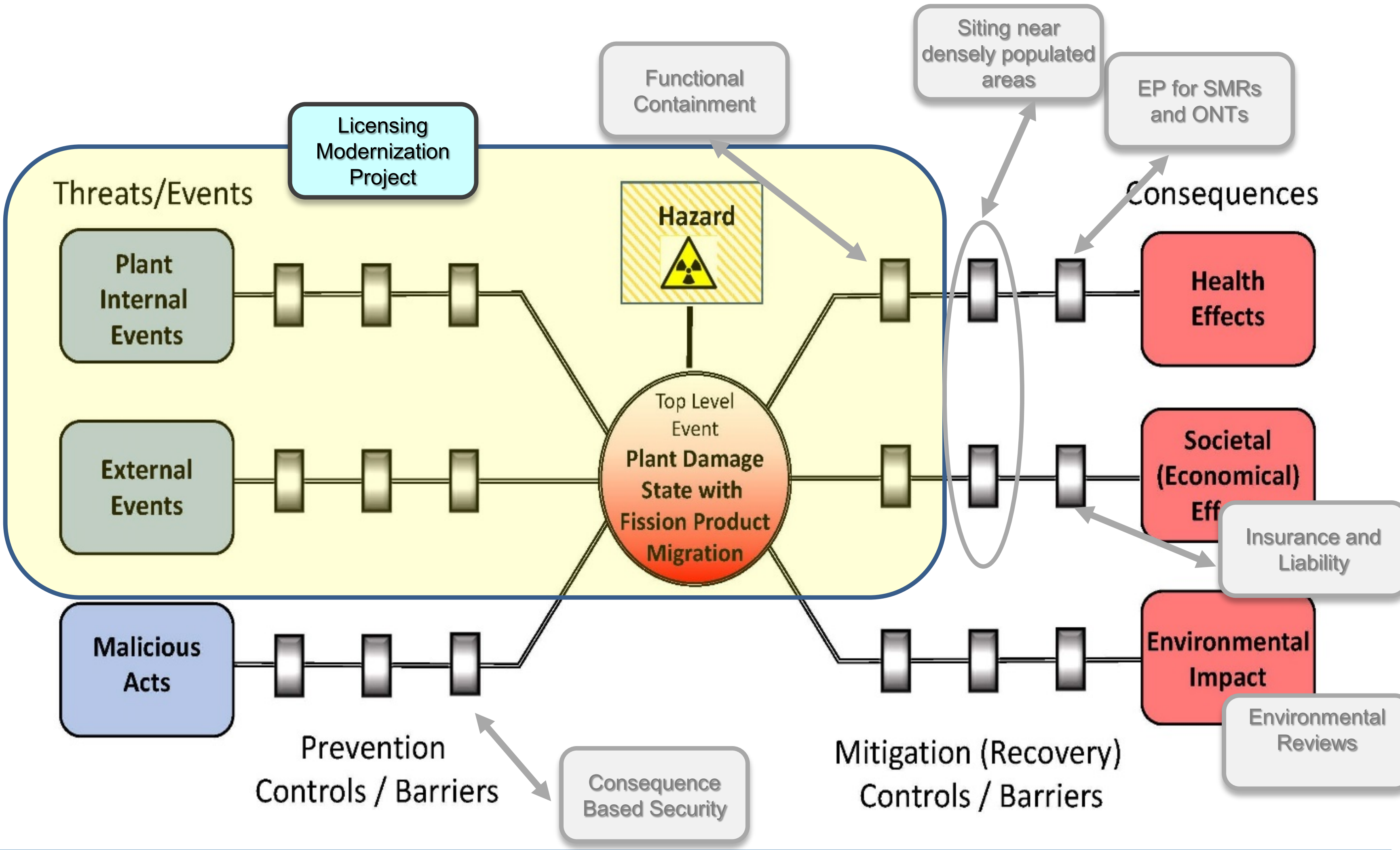
Figure 4.2. Definition of Risk-Significant and Safety-Significant SSCs



- F-C Target considered along with cumulative risk metrics, safety classification, and assessment of defense in depth



Integrated Approach



NRC Path Forward

- Draft a rulemaking plan, taking into consideration today's feedback
- Hold more focused public meeting(s) in the months ahead
- Finalize rulemaking plan and associated Commission paper
- Send rulemaking plan to Commission in April 2020.
- Documentation related to the Part 53 rulemaking can be found on the regulations.gov website by searching for the NRC Docket ID "**NRC-2019-0062**"

Nuclear Energy Innovation and Modernization Act: Establishing Metrics and Milestones

Steven Lynch
Acting Chief, Advanced Reactor Licensing Branch
Division of Advanced Reactors

Background

- Nuclear Energy Innovation and Modernization Act (NEIMA) signed into law on January 14, 2019
- Section 102(c) of NEIMA requires:
  1. The development of performance metrics and milestone schedules for “requested activities of the Commission”
  2. Reports for certain delays associated with these activities

Definition of "Requested Activities"

- Section 3(10) of NEIMA defines "requested activity of the Commission" as
  A. The processing of applications for
     i. Design certifications or approvals
     ii. Licenses
     iii. Permits
     iv. License amendments
     v. License renewals
     vi. Certificates of compliance
     vii. Power uprates
  B. Any other activity requested by a licensee or applicant
- In general, for the purposes of NEIMA, requested activities of the Commission involve the preparation and issuance of a final safety evaluation by the NRC

Establishing Milestone Schedules

- Generic milestone schedules for requested activities are provided on the NRC public webpage: https://www.nrc.gov/about-nrc/generic-schedules.html
  – Generic schedules based on historical data, ongoing reviews, and modernization efforts

- For the initial application reviews for non-light water reactors, a generic milestone schedule of 36 months (30 months if referencing a certified design) has been established

- Application-specific schedules, which may be shorter or longer, will be established for each requested review

Impacts on Specific Review Schedules

- Quality of application
  - Adherence to regulatory requirements
  - Technical completeness
  - Attention to detail (i.e., organization, format, etc.)

- Requests for additional information (RAIs)
  - Complexity and novelty of technology
  - Completeness, timeliness, and responsiveness to requests
  - Number of RAIs and need for follow-up
  - Evaluation of new information

- Policy questions
  - Commission involvement to resolve unique considerations

- Advisory Committee on Reactor Safeguards
  - Number of subcommittee meetings
  - Follow-up items

Other Scheduling Considerations

- Potential for contested hearing
- Mandatory hearing for certain applications
  - Cannot hold mandatory hearing until completion of Safety Evaluation Report, Environmental Impact Statement, ACRS Review, and any contested hearing
- Commission decision to issue or deny permit or license
  - Decisions typically made 2 – 4 months following mandatory hearing

Metrics and Reporting

- The performance indicator is 100 percent timely completion of final safety evaluations within the established generic milestone schedules
- The NRC staff to notify the Commission within 30 days of missing a generic milestone schedule
- The NRC staff to prepare a report to Congress if a requested activity is not completed within 180 days after the established generic milestone schedule

Performing Effective Reviews

- Meeting the performance metrics and milestone schedules established in NEIMA supports NRC staff commitment to performing effective reviews

- Prospective applicants should engage with NRC staff early on anticipated licensing actions to develop specific review schedules

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Nuclear Energy Innovation and Modernization Act (NEIMA): NRC Section 103 Activities

John Segala
Chief, Advanced Reactor Policy Branch
Division of Advanced Reactors
August 15, 2019

Sec. 103. Advanced Nuclear Reactor Program

a) Licensing
   1) **Staged Licensing**
   2) Risk Informed Licensing
   3) Research and Test Reactor Licensing
   4) Technology-Inclusive Regulatory Framework
   5) **Training and Expertise**
   6) Authorization of Appropriations

b) **Report to Establish Stages in Licensing Process**

c) **Report to Increase RIPB Techniques**

d) Report to Prepare RTR Licensing Process

e) Report to Complete Rulemaking

NRC Staff Activities

- Issued a Letter to Congress on July 12, 2019 (ADAMS Accession # ML19128A289) enclosing two reports:

1. Establishing Stages in Advanced Reactor Licensing (Sec. 103(b))
    - Implementation of stages in licensing process within 2 years
        - NRC has completed staged licensing activities with issuing Regulatory Review Roadmap
        - Topical Reports, Standard Design Approval, Preapplication Engagement, and Licensing Project Plans/Regulatory Engagement Plans
    - Required evaluations
        - Fuel Qualification, Industry Codes and Standards, etc.

NRC Staff Activities (Cont.)

2. Increasing Use of RIPB Techniques and Guidance (Sec. 103(c))
    - Licensing Modernization Project (NEI 18-04, DG-1353, and draft SECY paper)
    - Mechanistic Source Term (SECY-93-092, NRC contract with INL to develop guidance)
    - Containment Performance (SECY-18-0096 and RG 1.232)
    - Emergency Preparedness (Proposed rule provided to Commission on October 12, 2018)
    - Other Policy Issues (Siting as it relates to population, Physical Security, and Micro Reactors)

- Coordination and stakeholder input
  - Public Stakeholder meeting on March 28th
- Cost and schedule estimates
  - Non-fee recoverable advanced reactor appropriations

NRC Staff Activities (Cont.)

- Issued Internal Memo on August 8, 2019
  - Staff training or hiring of experts to support staged licensing, risk-informed licensing, research and test reactor licensing, and technology-inclusive regulatory framework (Sec. 103(a)(5))
    - Training (IAP Strategy 1)
      - Technology Training Courses (MSRs, SFRs, Micro, HTGRs)
      - Computer Code training (MOOSE and BISON)
      - Research and Test Reactor Training
    - Knowledge Management
      - Contractor Reports, MOUs with DOE to share expertise and knowledge, etc
    - Hiring
      - Competency Modeling, New Division, Core Review Team, and Merger of NRO and NRR

Non-LWR Vision and Strategy Computer Code Reports

Modeling and Simulation of 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

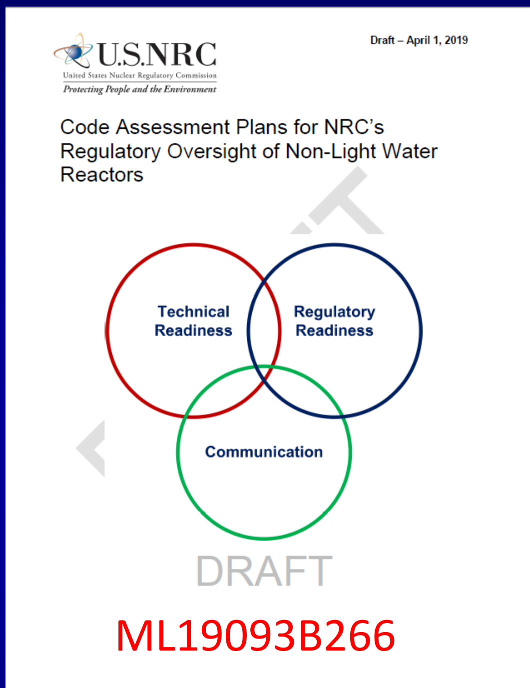
Advanced Reactor Stakeholder's Meeting
August 15, 2019

NRC's Integrated Action Plan (IAP) for Advanced Reactors

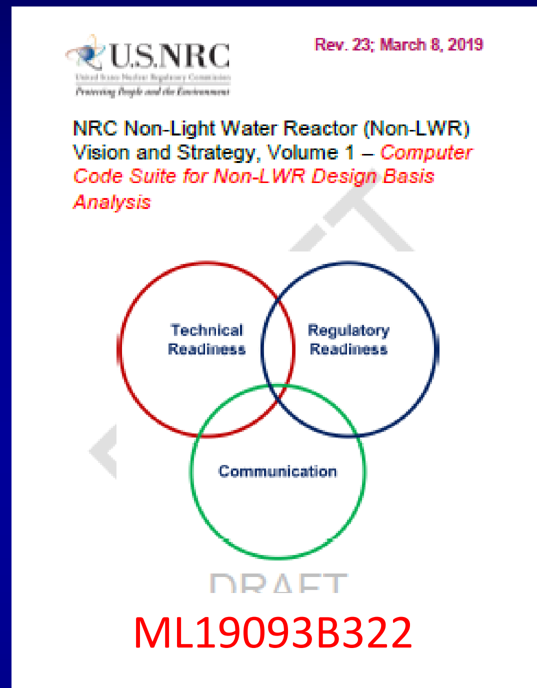
Near-Term Implementation Action Plan

- **Strategy 1** – Knowledge, Skills, and Capacity
- **Strategy 2** – Computer Codes
- **Strategy 3** – Flexible Review Process
- **Strategy 4** – Industry Codes and Standards
- **Strategy 5** – Technology Inclusive Issues
- **Strategy 6** – Communication

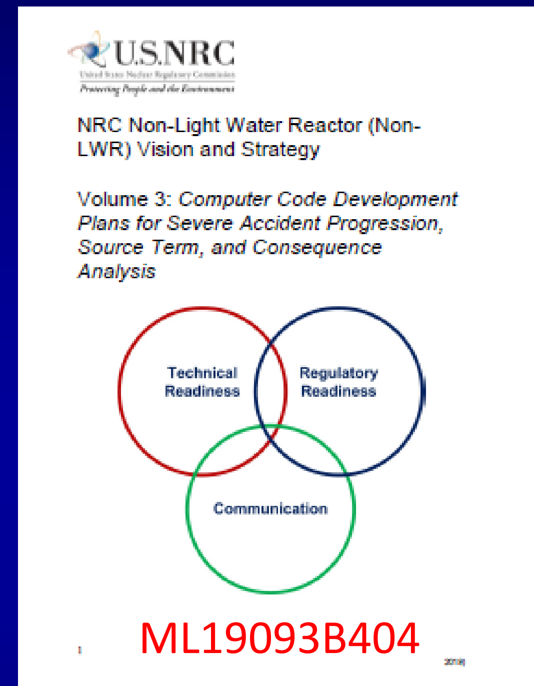
NRC's Implementation Action Plan, Strategy 2 – Computer Codes



Introduction



Volume 1



Volume 3

Volume 2 = Fuel Performance
Volume 4 = Radiation Protection

Under Development

Event Selection

- "Chapter 15" vs "Chapter 19" deterministic approach to be replaced with LMP.
- "Design Basis" Code(s) = those to be used for confirmatory analysis of events that little/no core (geometric) disruption or fission product release.
  - Unprotected loss of flow
  - Loss of heat sink(s)
  - Events that may involve multiple failures
- "Beyond Design Basis" Code(s) = for events involving core melt, fission product release & transport.

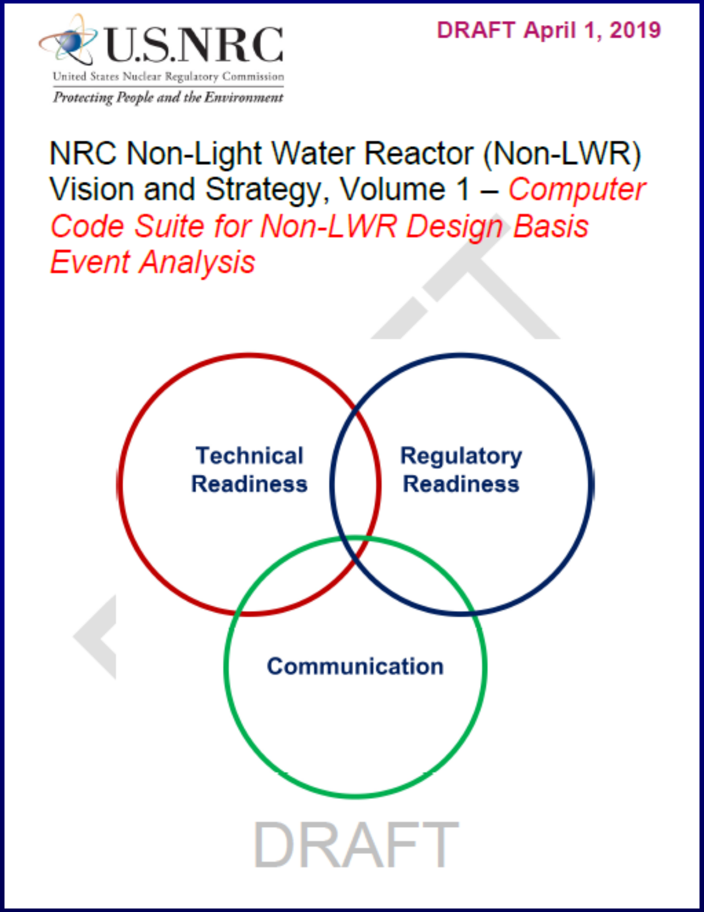
Some Recent & Upcoming Events . . .

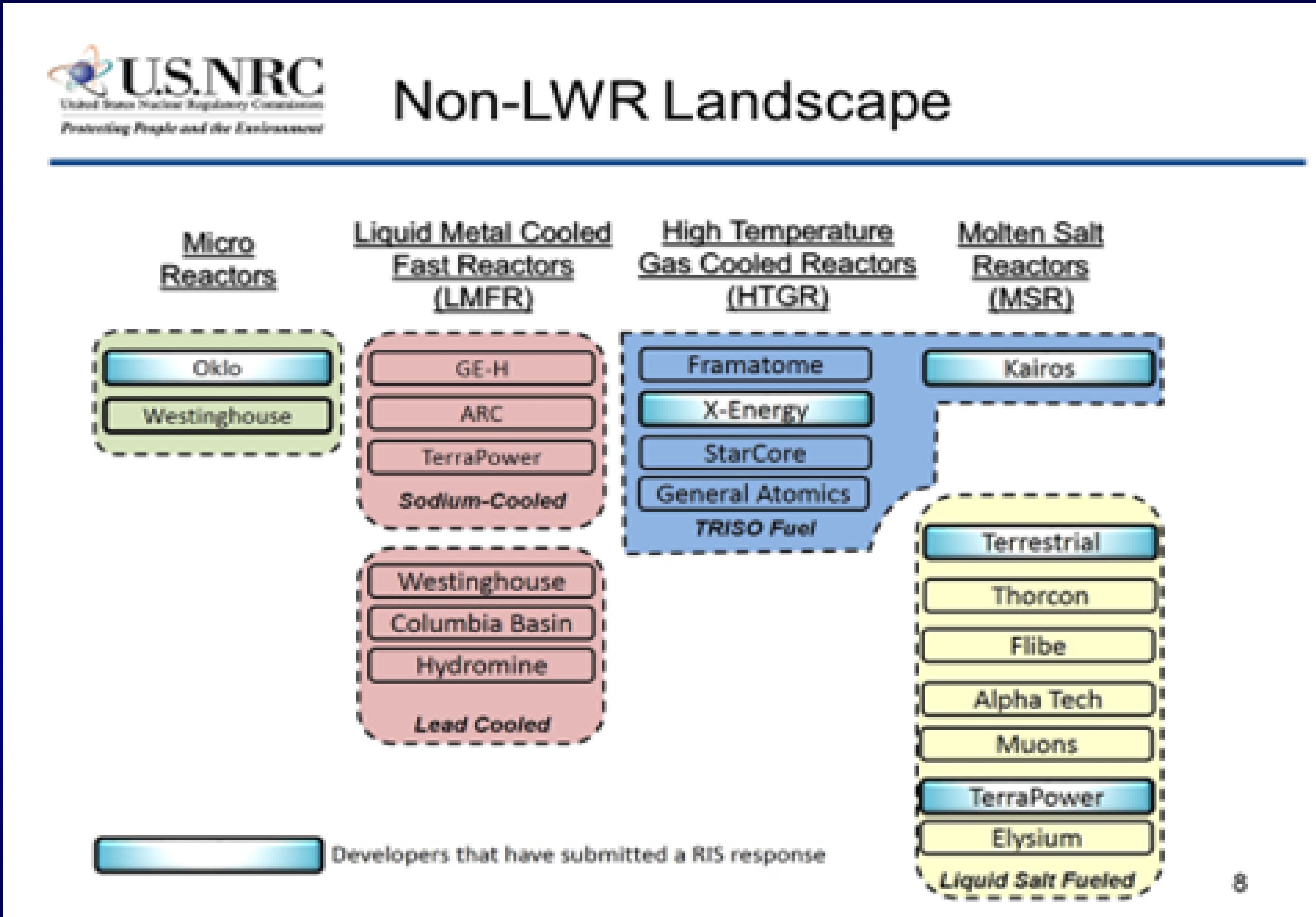
- Technical approach in Volumes 1 and 3 discussed with ACRS "Future Plant Design" subcommittee on May 1, 2019.
  - » Scenarios
  - » "Gaps"
- September 5 "Data Needs" Meeting with DOE.
- Volume 2 (Fuel Performance) to be discussed with ACRS "Future Plant Design" subcommittee on September 17, 2019.

Volume 1 “Design Basis Event Analysis”

Introduction / Outline

- Volume 1 “Design Basis Event Analysis” :
  - Phenomena Identification and Ranking Tables (PIRTs)
  - Event Scenarios
  - “New” Physical Phenomena for non-LWRs
  - Gaps
  - Tasks




Non-LWR Landscape
Micro Reactors
Oklo
Westinghouse
Liquid Metal Cooled Fast Reactors (LMFR)
GE-H
ARC
TerraPower
Sodium-Cooled
Westinghouse
Columbia Basin
Hydromine
Lead Cooled
High Temperature Gas Cooled Reactors (HTGR)
Framatome
X-Energy
StarCore
General Atomics
TRISO Fuel
Molten Salt Reactors (MSR)
Kairos
Terrestrial
Thorcon
Flibe
Alpha Tech
Muons
TerraPower
Elysium
Liquid Salt Fueled
Developers that have submitted a RIS response
8

Characterization of Design Types

Plant Type No.	Description	Fuel
1	HTGR; prismatic core, thermal spectrum	TRISO (rods or plates)
2	PBMR; pebble bed core, thermal spectrum	TRISO (pebbles)
3	GCFR; prismatic core, fast spectrum	SIC clad UC (plates)
4	SFR; sodium cooled, fast spectrum	Metallic (U-10Zr)
5	LMR; lead cooled, fast spectrum	Not available. (Possibly nitride fuel.)
6	HPR; heat pipe cooled, fast spectrum	Metallic (U-10Zr)
7	MSR; prismatic core, thermal spectrum	TRISO (plates)
8	MSPR; pebble bed, thermal spectrum	TRISO (pebbles)
9	MFSR; fluoride fuel salt, thermal/epithermal spectrum	Fuel salt
10	MCSR; chloride fuel salt, fast spectrum	Fuel salt

"Modeling Gaps" Identified by PIRTs

- Phenomena that are significant and "new" with increased importance for non-LWRs relative to conventional LWRs include but are not limited to:
  - Thermal stratification and thermal striping
  - Thermo-mechanical expansion and effect on reactivity
  - Large neutron mean-free path length in fast reactors
  - Transport of neutron pre-cursors (in fuel salt MSRs)
  - Solidification and plate-out (MSRs)
  - 3D conduction / radiation (passive decay heat removal)

↓

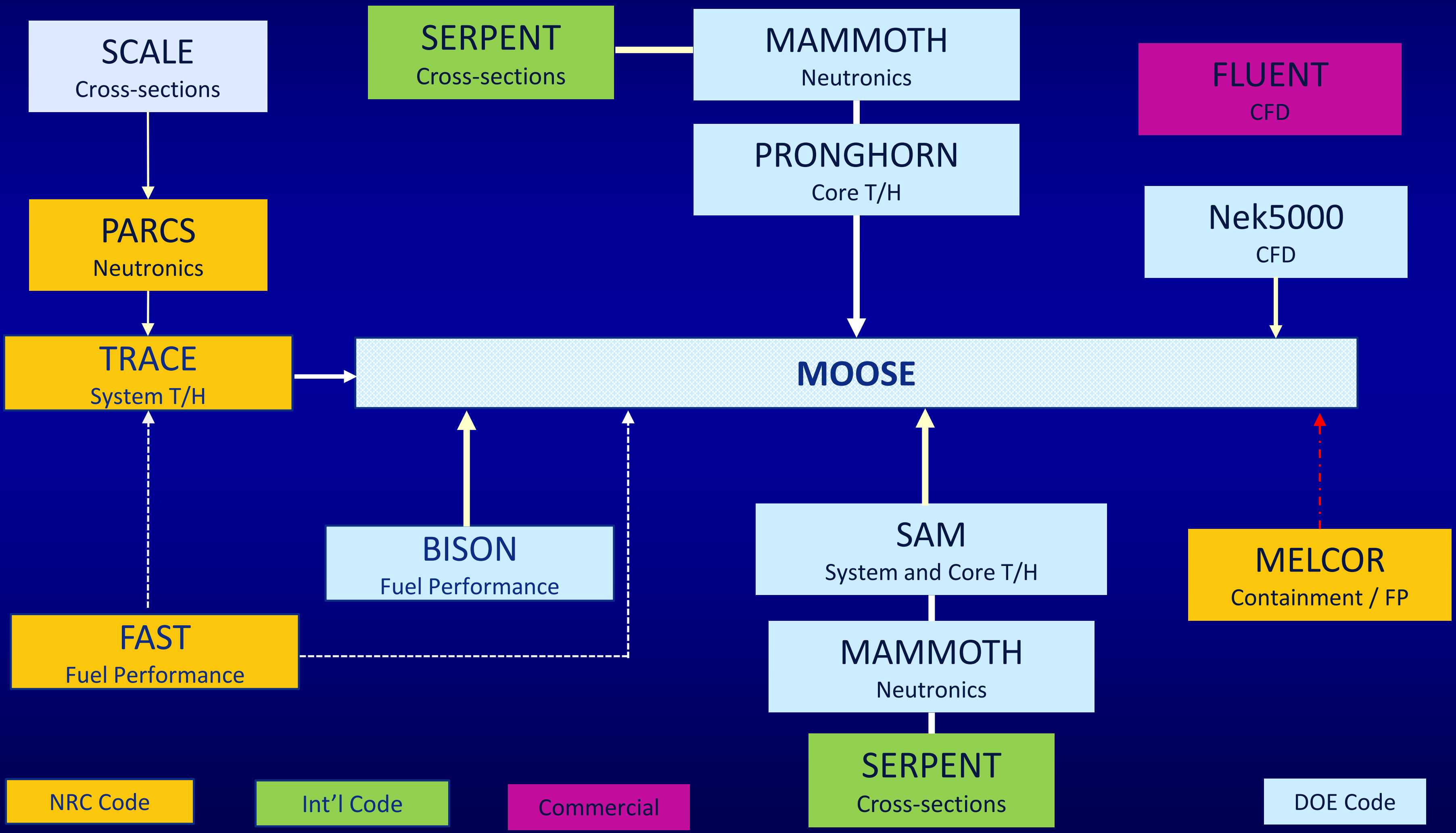
"Modeling Gaps in NRC Codes"

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 validation is critical and represents the major gap in EM development. Database is weak for some designs.
- Computation Requirements. Must be able to run simulations on NRC desktops or HPC platforms readily available to NRC.

Codes selected for CRAB satisfy these criteria.

Comprehensive Reactor Analysis Bundle (CRAB)



Approach to Validation

(1) Review PIRT phenomena & prior test programs for applicability to each of the new designs.

(2) Identify and prioritize validation tests (based on PIRT findings and NRO expected review schedule).

(3) Develop “reference plant” models to define nodalization scheme and modeling options.

(4) Coordinate efforts with DOE and national labs to complete validation & improve code performance based on findings.

Summary & Conclusions

- **"Volume 1" recommends the codes in CRAB as the approach for non-LWR DBE analysis. Flexibility to simulate multiple designs (including LWRs with ATF).**
- **"Gaps" in code capability, V&V are identified along with tasks for resolution.**
- **Using the combination of NRC and DOE codes will provide a <u>technically superior product</u> than can be attained with further development of only the NRC's conventional LWR codes.**

Extra Stuff

Codes for Design Basis Event Analysis

- Codes considered:
  - NRC codes (TRACE, PARCS, FRAPCON, FAST)
  - DOE NEAMS codes (MAMMOTH, PRONGHORN, RELAP7)
  - ANL codes (SAS4A/SASSY, SAM, PROTEUS, 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 both NRC and DOE.

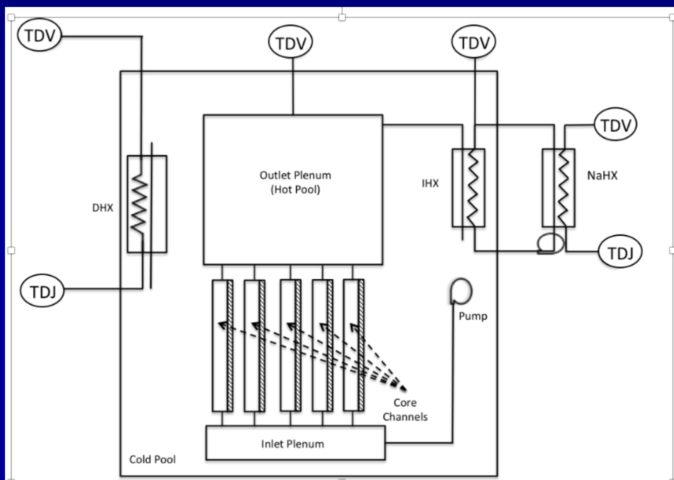
Unique Capabilities Available in CRAB

- Examples
  - Multiphysics Coupling
  - Geometric Fidelity
  - Advanced Equivalence Methods
  - Multi-Scheme Capability
  - 3D Reduced Order Flow Model

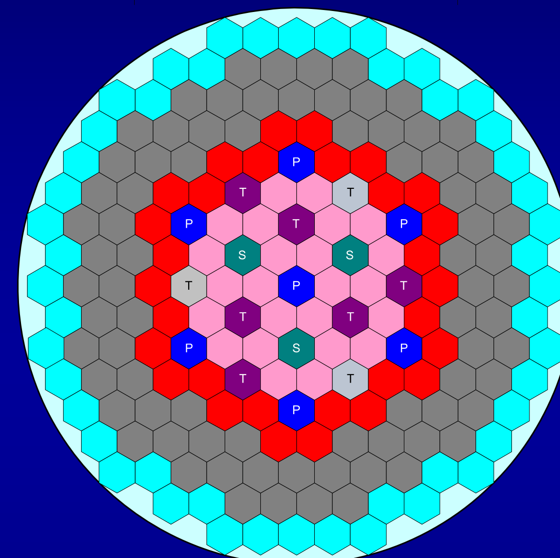
Goal: Enable analysis of advanced designs without over-simplifying assumptions to provide intermediate fidelity model for modest computational resources.

Multiphysics Coupling

SAM: System Level Thermo-Fluids



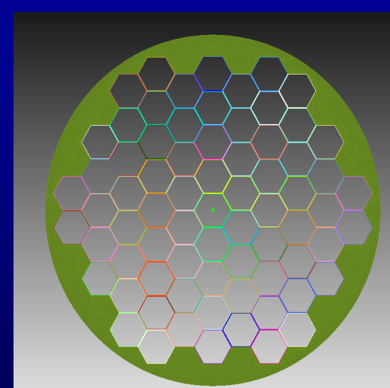
MAMMOTH: Rx Kinetics



Temperatures & Densities

Power

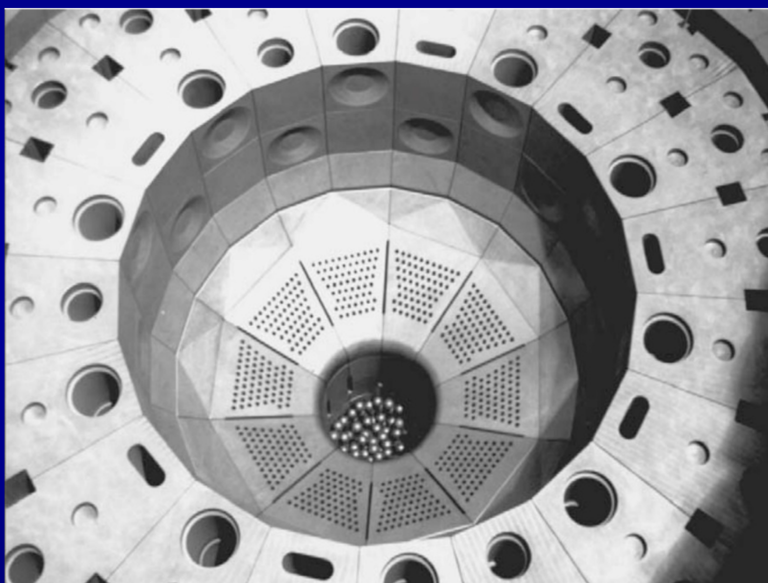
Temperatures



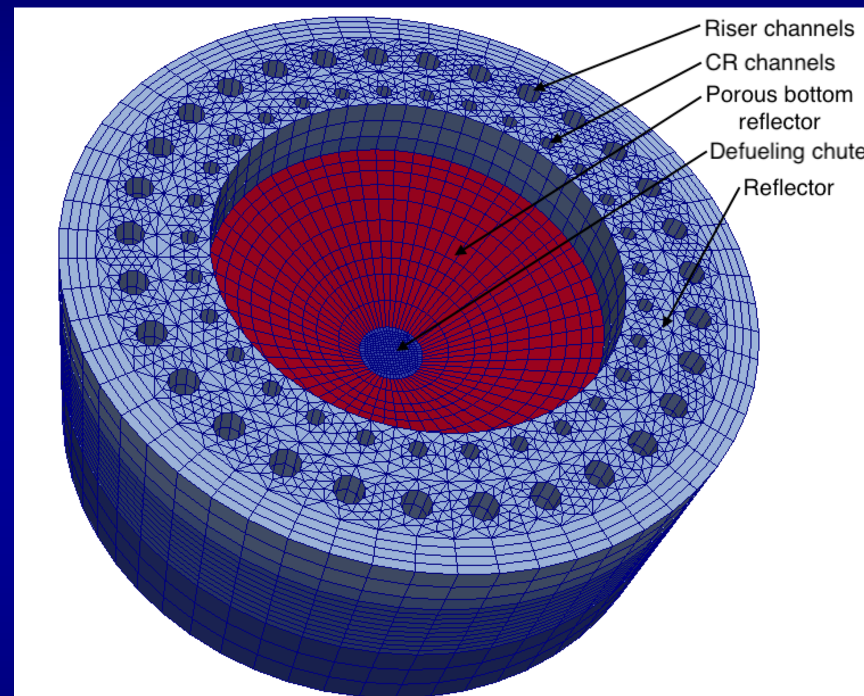
Displacements

Tensor Mechanics Module

Geometric Fidelity



HTR-10
Bottom Reflector & Conus

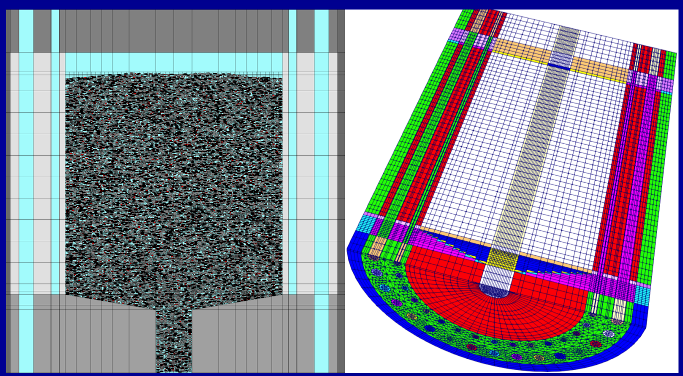


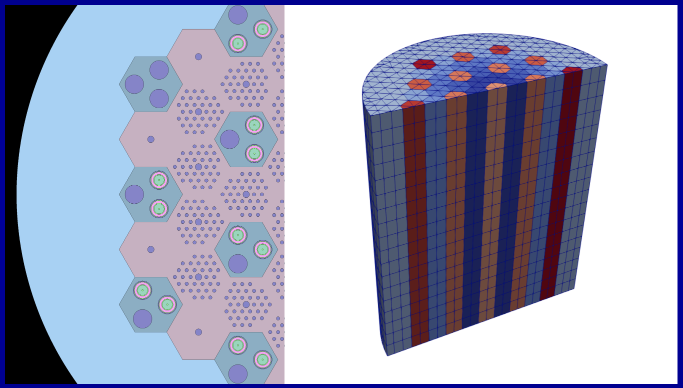
PRONGHORN Mesh
- 1-D Channels for upcomer & CR's
- 3-D Porous body for conus

Advanced Equivalence Methods

- **HTR-PM**
  - Pebble bed HTR (420k pebbles)
  - Diffusion problem size – 54,656 cells

- **HTTR**
  - Prismatic HTR
  - Diffusion problem size – 15,552 cells





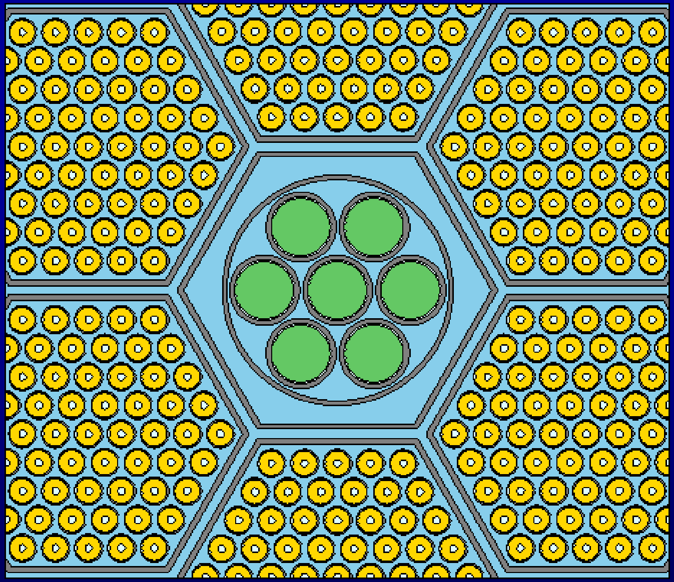
HTR-PM (T_{fuel} = 1100 K)	k_{eff}	Δpcm	RMS % Err $\nu\Sigma_f$	Max %Err $\nu\Sigma_f$
Serpent	1.01159	±1.8	-	-
Diffusion	1.03653	2435	6.0	40.6
SPH-Diffusion	1.01159	0	1.55E-03	2.94E-03

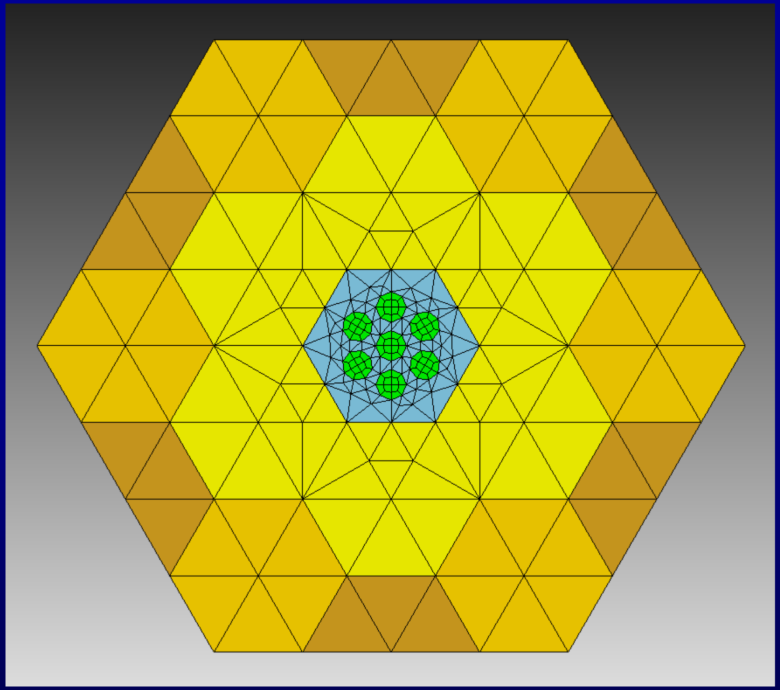
HTTR (T_{fuel} = 1300 K)	k_{eff}	Δpcm	RMS % Err Power	Max %Err Power
Serpent	1.00259	±2.7	-	-
Diffusion	1.01978	1715	3.12	6.20
SPH-Diffusion	1.00259	0	7.0E-02	2.0E-01

Transport level accuracy for the price of a diffusion calculation

Multi-Scheme Capability

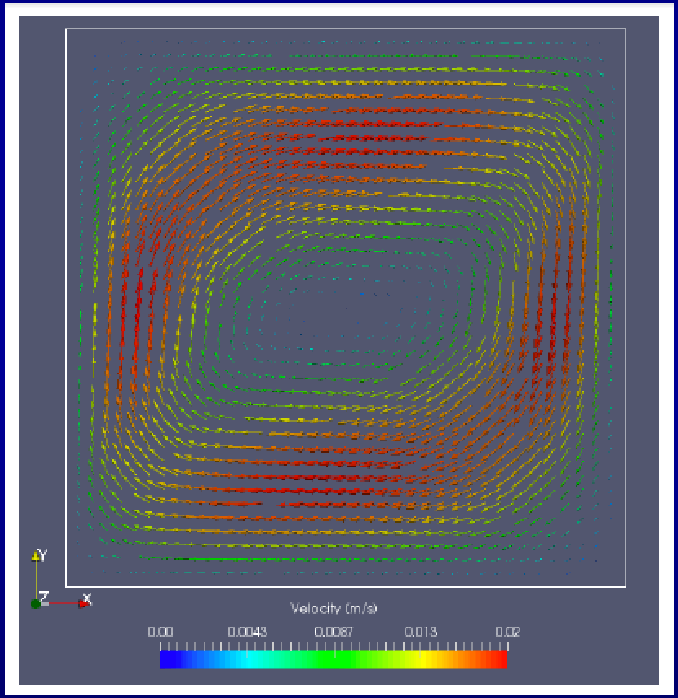
- MAMMOTH
  - Allows usage of transport where more detail is needed with efficiency of diffusion for remainder of domain



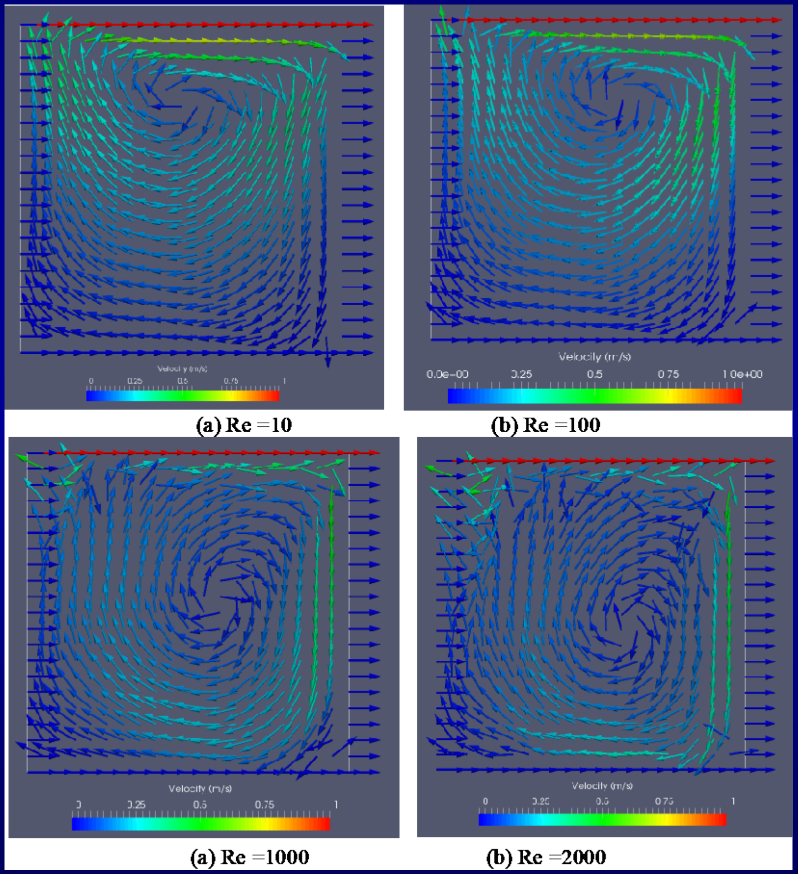


SAM: 3-D Flow Model

- Validation Examples



Natural Convection
Ra = 10^5



Lid-Driven Cavity Flow

Verification & Validation "Gaps"

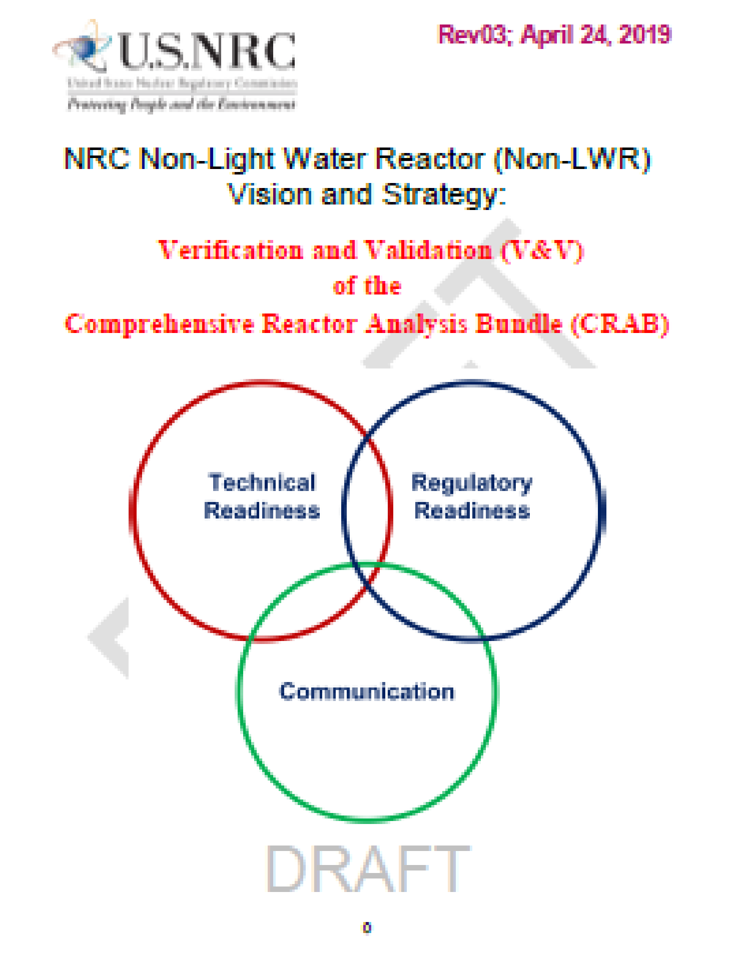
Code Assessment Issues

- Code "Assessment" = Verification & Validation represents the most significant "gap" in readiness for the DBE analysis codes.
- Verification: Considered generally good – however "coupling" may need additional cases to ensure conservation of mass, energy & momentum.
- Validation: Completed and on-going validation shows good agreement between predicted & measured results. More is needed, and should be done with a "frozen" code.

Code Validation Matrix

- Volume 1 identifies the most important validation cases for each of the 10 design types. Additional validation is being performed by DOE as part of developmental assessment.

- An additional report is being developed to summarize all of the V&V needed for CRAB.


U.S.NRC

Rev03; April 24, 2019

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy:

Verification and Validation (V&V) of the Comprehensive Reactor Analysis Bundle (CRAB)

DRAFT

0

Validation Status

GCRs: HTR-10, PBMR-268,-400, SANA, HTTU, AVR, . . .

SFRs: EBR-II, FFTF, CEFR, ZPPR, Monju, . . .

LMRs: Helios

HPRs: KRUSTY, Godiva

MSRs: MSRE, UCB-Ciet, UW-Loop, . . .

RCCS: NSTF, UW-Loop, . . .

Completed
In-progress
Planned

- There are significant "gaps" : Validation is partial, with numerous tests in-progress or planned.
- More importantly, there is a lack of experimental data for some designs.

Validation / Experimental "Gaps"

GCRs: Prismatic gas-cooled IET (i.e. HTTR, OSU-HTTF)

SFRs: Pool type IET data, International data

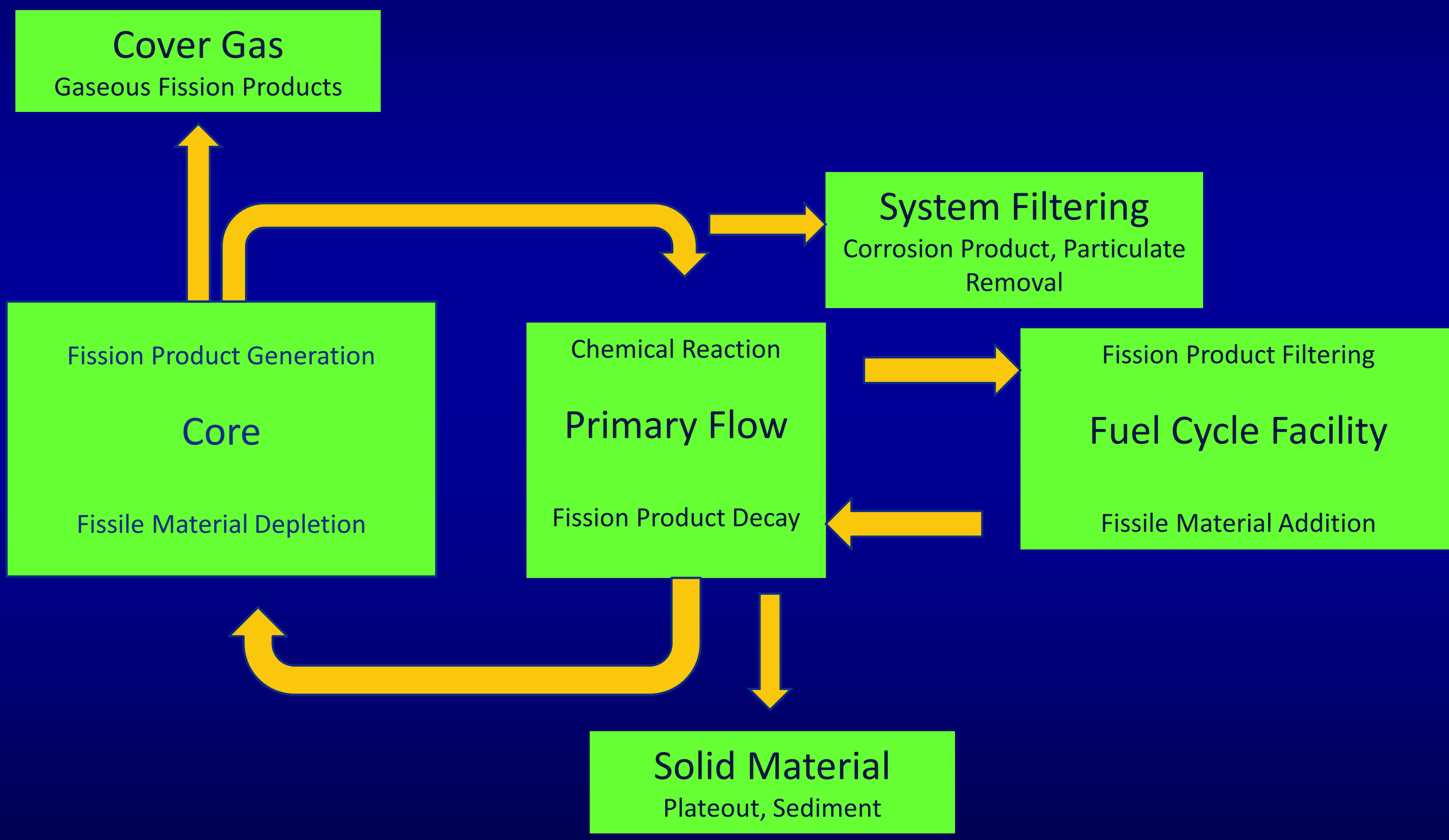
LMRs: Additional IET data, SET data for T/H, fuel, kinetics

HPRs: Monolith conduction and heat release SET data

MSRs: Pool type IET data, natural circulation loop data

Scaling of IETs and Range of Conditions of existing data to full-scale prototypes remains to be established.

Molten Salt Reactor (Inventory Control “Gap”)



NRC non-LWR COMPUTER CODE DEVELOPMENT PLANS FOR SEVERE ACCIDENT PROGRESSION, SOURCE TERM, AND CONSEQUENCE ANALYSIS

https://www.nrc.gov/docs/ML1914/ML19143A120.pdf

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Future Plant Designs Subcommittee
Docket Number:	(n/a)
Location:	Rockville, Maryland
Date:	Wednesday, May 1, 2019
Work Order No.:	NRC-0327 Pages 1-332

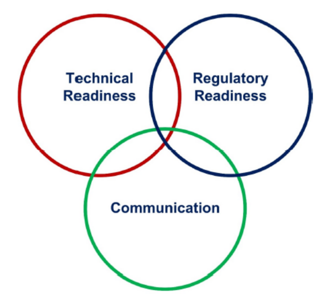
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

August 15, 2019

ML19093B404

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy

Volume 3: *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*



DRAFT - April 1, 2019

Design Basis Source Term Development Process

(example: MOX & High Burnup Fuel)

Experimental Basis ← PIRT process

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

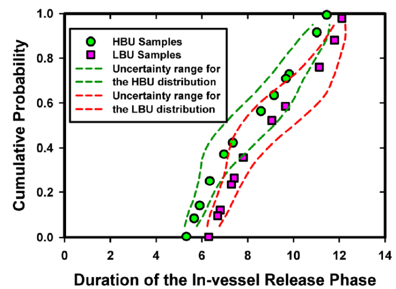
MELCOR

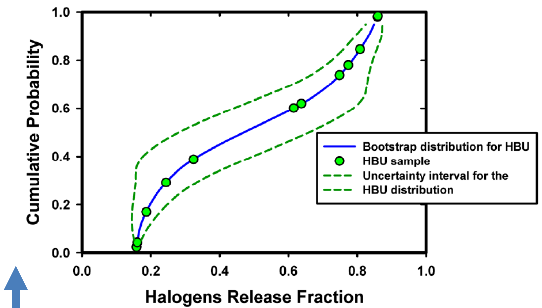
Accident Analysis

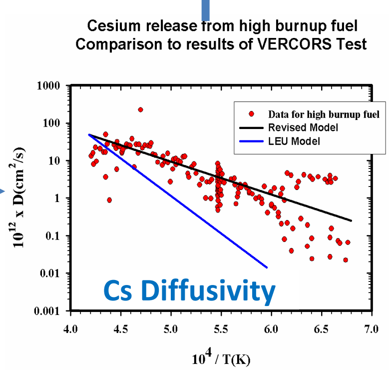
Scenario # 1	Scenario # 2
………………	………………
Scenario # n-1	Scenario # n

Synthesize timings and release fractions

Design Basis Source Term





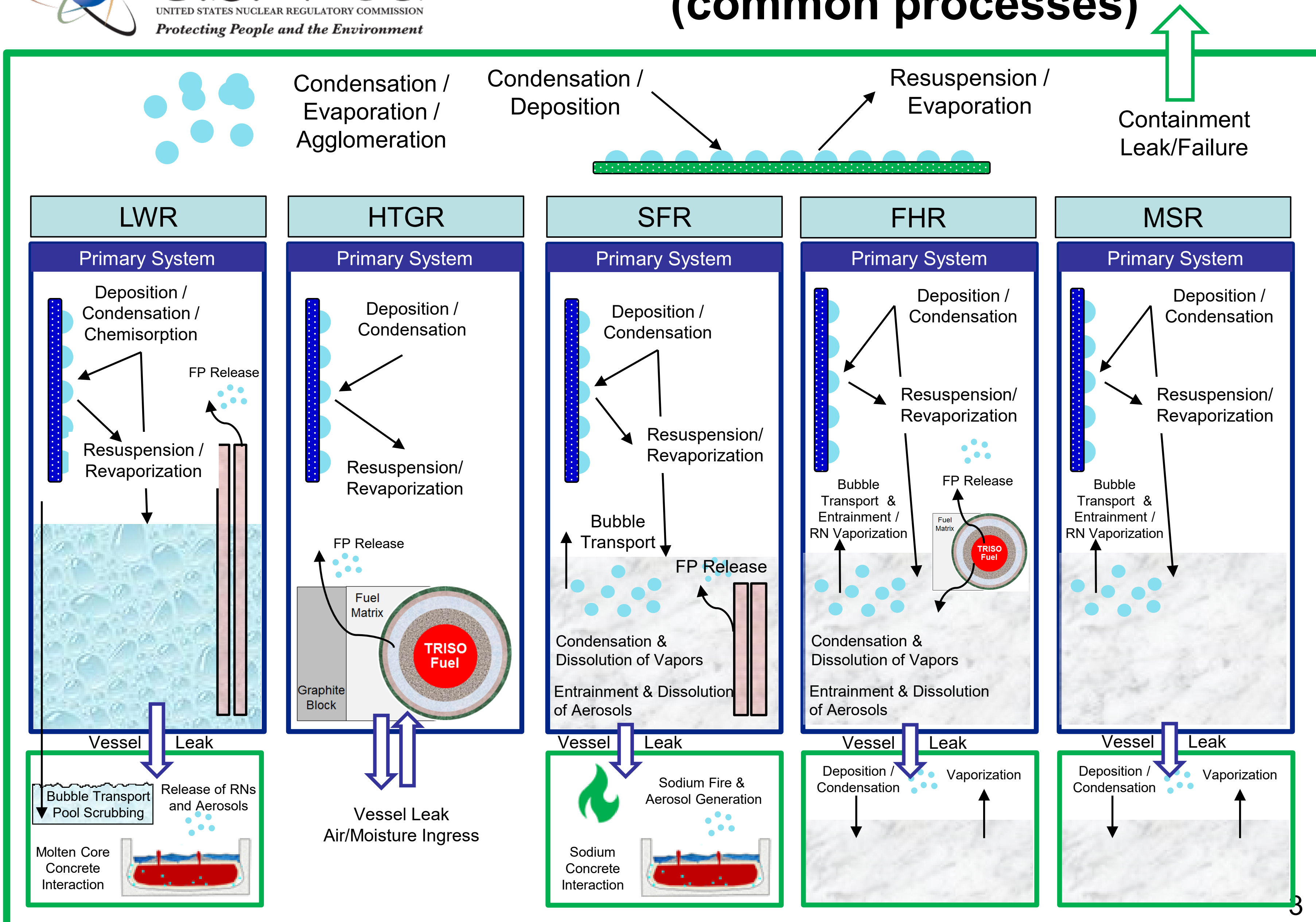


Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	Late 10 hr
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0

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

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

Phenomenology & Release Paths (common processes)



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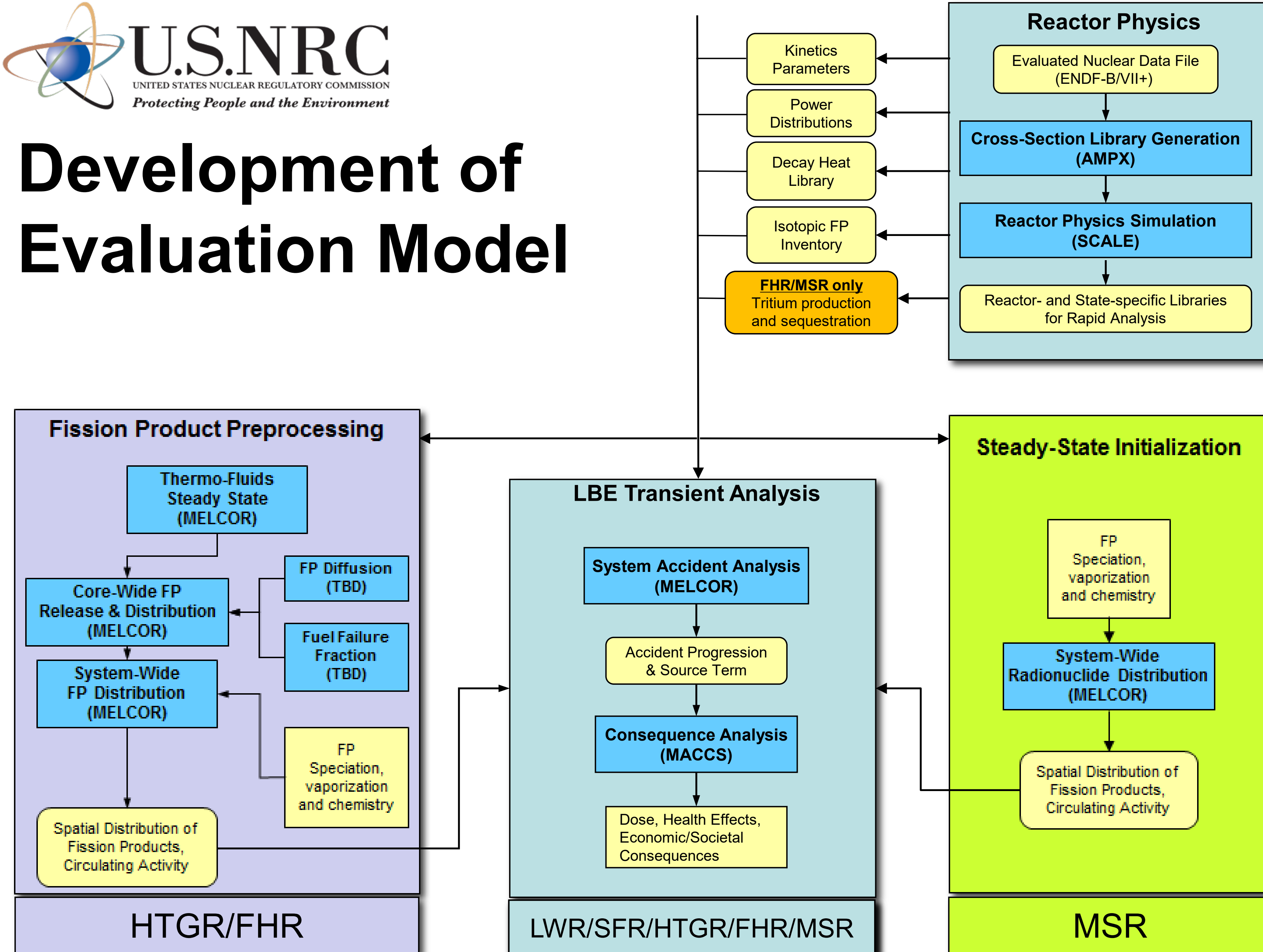
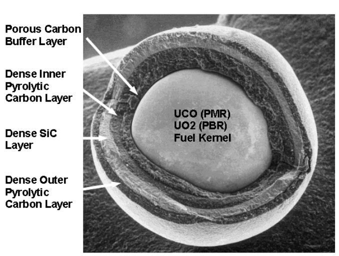
Development of Evaluation Model



MELCOR Input & Data Requirements

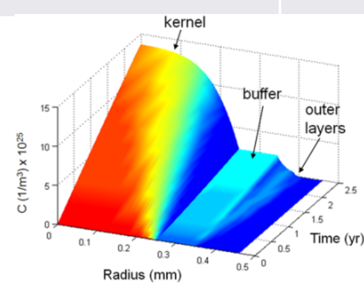
Input Data	HTGR	SFR	MSR	FHR
FP Inventory	SCALE	SCALE	SCALE	SCALE
FP diffusion coefficients (D) and release	Experiments (e.g., AGR) and analysis (e.g., DOE tools)	Experiments		Experiments (e.g., AGR) and analysis (e.g., DOE tools)
Core power shape	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)
Fuel failure	Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)		Experiments/other codes (e.g., DOE tools)
Dust generation & FP transport	Experiments, historical data and other code (e.g., DOE tools)			
FP release under air/water ingress & interaction w/ graphite	Experiments			
Kinetics parameters and reactivity feedback coefficients	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)
Equilibrium constants for release from pool and vapor pressure data		Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)



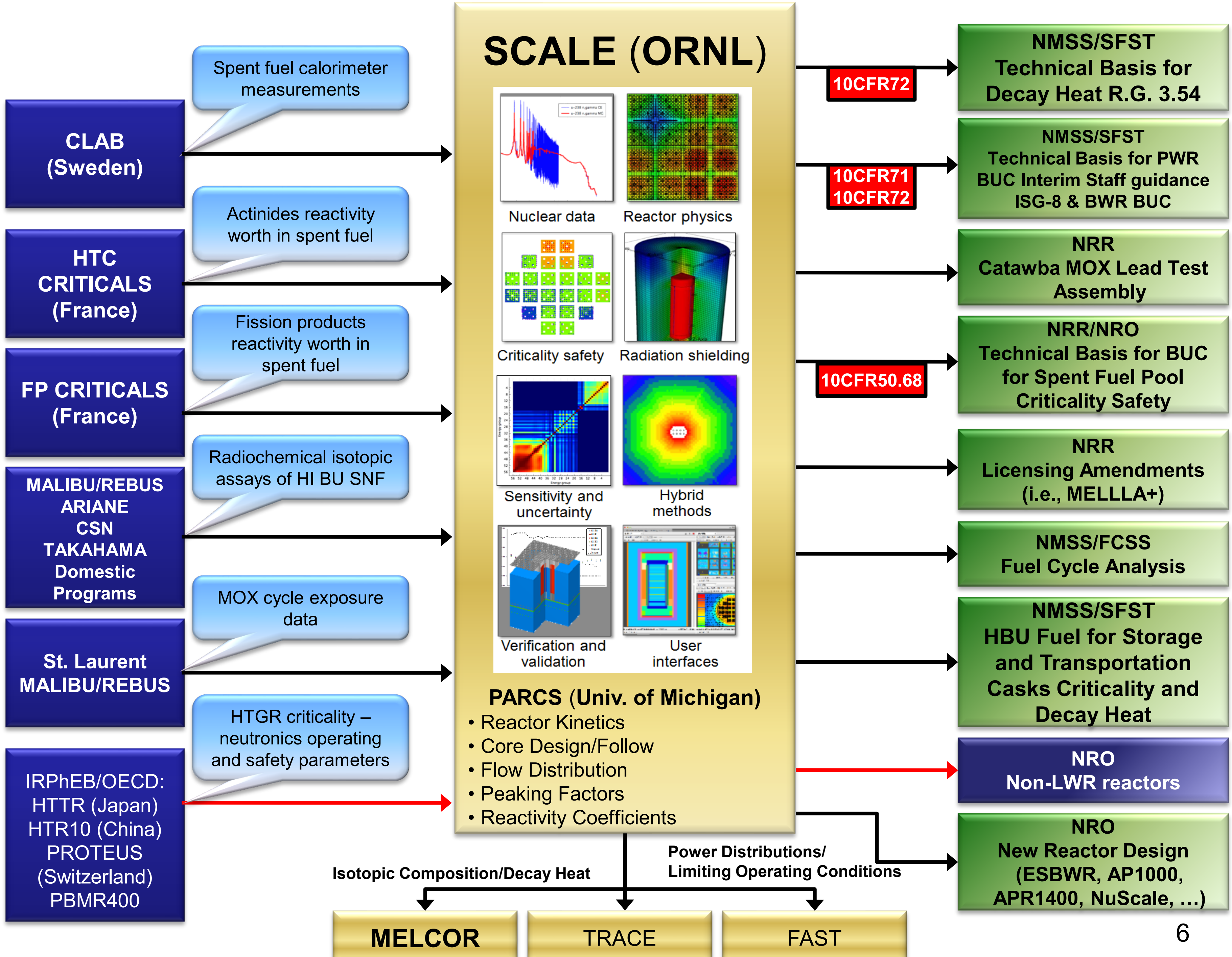
$$\frac{\partial C}{\partial t}=\frac{1}{r^m}\frac{\partial}{\partial r}\left(r^m D \frac{\partial C}{\partial r}\right)-\lambda C+S$$

SCALE

Experiments/Analysis



Distribution of Cs-137 in different layers as a function of time


SCALE (ORNL)
Nuclear data
Reactor physics
Criticality safety
Radiation shielding
Sensitivity and uncertainty
Hybrid methods
Verification and validation
User interfaces
PARCS (Univ. of Michigan)
• Reactor Kinetics
• Core Design/Follow
• Flow Distribution
• Peaking Factors
• Reactivity Coefficients
CLAB (Sweden)
Spent fuel calorimeter measurements
HTC CRITICALS (France)
Actinides reactivity worth in spent fuel
FP CRITICALS (France)
Fission products reactivity worth in spent fuel
MALIBU/REBUS ARIANE CSN TAKAHAMA Domestic Programs
Radiochemical isotopic assays of HI BU SNF
St. Laurent MALIBU/REBUS
MOX cycle exposure data
IRPhEB/OECD: HTTR (Japan) HTR10 (China) PROTEUS (Switzerland) PBMR400
HTGR criticality – neutronics operating and safety parameters
10CFR72
NMSS/SFST Technical Basis for Decay Heat R.G. 3.54
10CFR71 10CFR72
NMSS/SFST Technical Basis for PWR BUC Interim Staff guidance ISG-8 & BWR BUC
NRR Catawba MOX Lead Test Assembly
10CFR50.68
NRR/NRO Technical Basis for BUC for Spent Fuel Pool Criticality Safety
NRR Licensing Amendments (i.e., MELLLA+)
NMSS/FCSS Fuel Cycle Analysis
NMSS/SFST HBU Fuel for Storage and Transportation Casks Criticality and Decay Heat
NRO Non-LWR reactors
NRO New Reactor Design (ESBWR, AP1000, APR1400, NuScale, …)
Isotopic Composition/Decay Heat
Power Distributions/ Limiting Operating Conditions
MELCOR
TRACE
FAST

SCALE Development

- Leveraging decades of physics models, nuclear data, and validation that can be extended to non-LWRs
  - Most efficient approach to support accident progression and source term analysis
  - For some technologies, the models are ready to be tested
- Experimental Needs
  - Decay heat, isotopic, validation data consistent with design and expected operating envelope
  - Criticality benchmarks
  - Destructive assay data for new fuel forms (e.g.: TRISO)
- Capabilities will be enhanced as more experience is gained, and gaps and uncertainties are quantified
- Plan will be updated as more experience is gained and as new information regarding specific reactor design becomes available. Current focus on:
  - How data transfer will work between SCALE and MELCOR/MACCS
  - Moving fuel and power history presents challenges
    - Demonstrate the sufficiency of bounding analysis for licensing use
  - Correct level of chemistry modelling between SCALE and MELCOR

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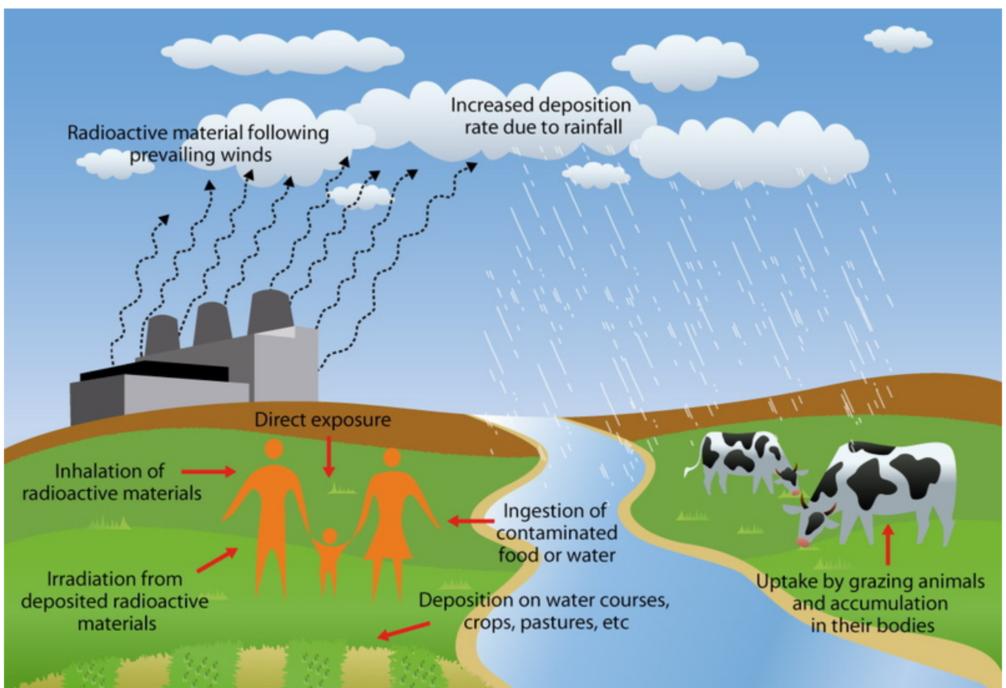
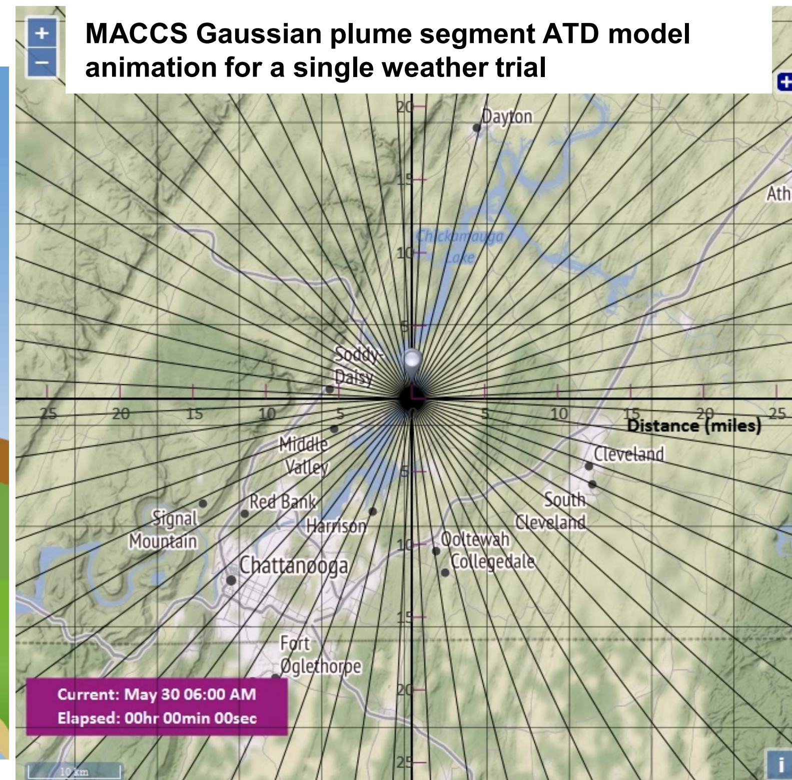
MACCS Overview

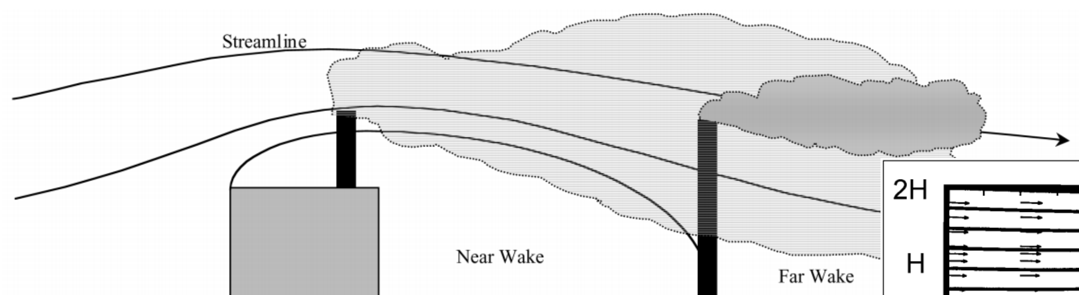
- MACCS is the only code used in U.S. for probabilistic offsite consequence analysis
- Highly flexible code that treats all technical elements of Level 3 PRA standard: radionuclide release, atmospheric transport, meteorology, protective actions, site data, dosimetry, health effects, economic factors,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



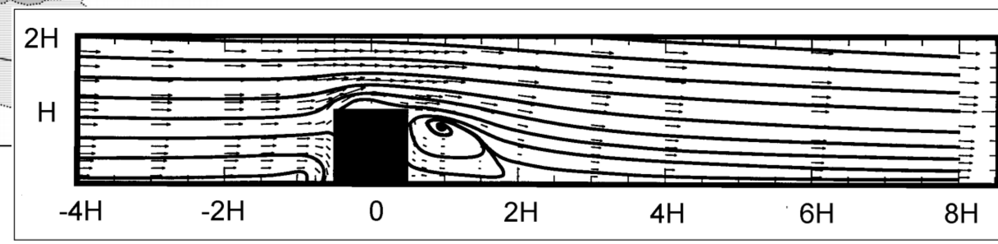


MACCS Code Development Areas for Non-LWRs

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



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



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

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Protecting Sensitive Information

Advanced Reactors Stakeholder Meeting – August 15, 2019

Stu Magruder
Senior Project Manager
Advanced Reactor Licensing Branch
Division of Advanced Reactors

Protecting Information

- NRC must protect classified and sensitive information
  - Classified information
  - Safeguards Information
  - Sensitive unclassified non-safeguards information SUNSI (e.g., proprietary, security-related, export controlled information (ECI))

Note: The NRC does not designate ECI.

ECI designation should be coordinated with appropriate federal agency (e.g., Department of Energy, Department of Commerce).

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SUNSI Categories

- Allegation information
- Investigation information
- Critical Electric Infrastructure Information (CEII)
- Security-related information
- Proprietary information
- Privacy Act information
- Federal-, State-, foreign government-, and international agency-controlled information (ECI)
- Sensitive internal information

Requests for Withholding

- Per 10 CFR 2.390, prospective applicants may request that proprietary information be withheld from public disclosure
- Requests for withholding must be accompanied by an affidavit
  - Affidavit should be either notarized or signed under oath or affirmation
  - Identify what information is considered proprietary
  - Explain why the release of information would cause harm
- Sensitive information, including proprietary information and ECI should include appropriate portion and page markings
- Non-proprietary (public) versions of documents should be provided with proprietary submittals
- NRC staff will evaluate requests and determine whether information should be withheld from public disclosure

Marking Guidance

MARKING	
What documents should be marked?	•Mark all documents containing Trade Secrets or Confidential Commercial or Financial Information. •Do not mark documents from INPO designated INPO Private.
Who may authorize document marking?	NRC recipient or originator (or supervisor) pursuant to 10 CFR 2.390.
How should a document be marked?	**NRC Generated Documents**: •The top and bottom of each page should be marked -**"Official Use Only – Proprietary Information."** **Incoming Documents**: •Marking requirements are defined in 10 CFR 2.390(b) and require marking only at the top of page, and each successive page containing proprietary Information, and adjacent to the specific proprietary information.
When is portion or page marking required?	•Required for all documents. •If the entire page is not affected, indicate the basis (i.e., trade secret, etc.) for the designation adjacent to the protected information. See 10 CFR 2.390 (b)(1)(i)(B).

Staff Guidance

- NRC Office Instruction LIC-204, "Handling Requests to Withhold Proprietary Information from Public Disclosure." (ADAMS Accession No. ML093240489)
  - Provides specific information on reviewing and dispositioning requests to withhold proprietary information
  - Publicly available – recommend reviewing before submitting documents

Additional Thoughts

- Take care with redacting process
- Expectations for level of detail in withholding may change with maturity of application
- What about requests for withholding information during public meetings?
- “No comment” policy for staff re SUNSI

Why is this Important?

- Final NRC records and documents are generally made public per 10 CFR 2.390(a)
  - Balance interests of industry and public
- Freedom of Information Act (FOIA) requests covered by 10 CFR Part 9 Subpart A
  - Documentation for withholding is important

Questions?

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