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Presentations for March 22, 2017 Public Meeting on Regulatory Process Improvements for Advanced Reactor Designs

1) NRC Staff Presentation
   - Mid- and Long-Term Implementation Action Plans
   - Security Design Considerations
   - Licensing Project Plans
   - Licensing Basis Events (for afternoon session)
2) Nuclear Infrastructure Council Presentation
   - Policy Issues
3) Nuclear Innovation Alliance Presentation
   - Major Portions Discussion for Standard Design Approval
4) Nuclear Energy Institute Presentation
   - Licensing Technical Requirements Modernization Project (Licensing Basis Events)

Advanced Reactor Program

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Public Meeting on Possible Regulatory Process Improvements for Advanced Reactor Designs

March 22, 2017

Advanced Reactor Program

Public Meeting

- Telephone Bridge

  (888) 603-9622

  Passcode: 6735363
- Opportunities for public comments and questions at designated times
- Please mute phones

Outline

- Introduction
- Joint Discussions
  - Mid- and Long-Term Implementation Action Plans
  - Security Design Considerations
  - Licensing Project Plans
- Policy Issues Discussion (NIC)
- Major Portions Discussion (NIA)
- Next Meetings/Public Discussion
- Licensing Technical Requirements Modernization Project (NEI/Southern)
- Public Questions/Comments

Mid-Term IAPs

1. Continue to acquire/develop sufficient technical skills and capacity to perform regulatory reviews and to conduct oversight of non-LWRs.

2. Continue to acquire/develop sufficient computer codes and tools to perform non-LWR regulatory reviews.

3. (a) Continue to develop guidance for a flexible non-LWR regulatory review process within the bounds of existing regulations, including the use of conceptual design reviews and staged-review processes.
   (b) Initiate and develop a new non-LWR regulatory framework (if needed) that is risk-informed, performance-based, and that features staff review efforts commensurate with the risks posed by the non-LWR NPP design being considered.

Mid-Term IAPs

4 Continue to facilitate industry codes and standards needed to support the non-LWR life cycle (including fuels and materials).

5 Identify and resolve technology-specific policy issues that impact the regulatory reviews, siting, permitting, and/or licensing of non-LWR NPPs.

Long-Term IAP

3

Continue to develop, finalize, and promulgate a new non-LWR regulatory framework (if needed) that is risk-informed, performance-based, and that features staff review efforts commensurate with the risks posed by the non-LWR NPP design being considered.

Security Design Considerations

- 10 CFR 73.55 – Physical security requirements
- 10 CFR 73.54 – Cyber security requirements
- Policy Statement on the Regulation of Advanced Reactors (73 FR 60612; October 14, 2008)
- "Security design considerations" should be considered while developing the facility design to ensure nuclear power reactors are protected against the design basis threat for radiological sabotage.
- Draft security design considerations (ML16305A328) issued for public comment (FRN 2017-04873; March 13, 2017)
- Regulations.gov docket number NRC-2017-0073
- Comments requested by April 27, 2017

Security Design Considerations

(1) Intrusion detection systems.

The design of physical security structures, systems, and components relied on for interior and exterior intrusion detection functions should provide assurance of detecting unauthorized access into vital and protected areas. The design should apply the principle of diversity necessary for the reliability and availability of systems and components to achieve the intended intrusion detection functions.

(2) Intrusion assessment systems.

The design of physical security structures, systems, and components relied on for intrusion assessment functions should provide assurance of rapid remote assessment for determining cause and initiating appropriate security responses. The design should apply the principle of diversity necessary for the reliability and availability of systems and components to achieve the intended intrusion assessment functions.

Security Design Considerations

(3) Security communication systems.

The design of structures, systems, and components (e.g., dedicated or plant operations systems) relied on for security communications should provide assurance of continuity and integrity of communications. Communication systems should account for design basis threats that can interrupt or interfere with continuity or integrity of communications. The design should apply the principles of redundancy and diversity.

(4) Security delay systems.

The design of structures, systems, and components relied on for delay functions should provide assurance for security responses to adversary attacks. The design of security delay systems should be appropriately layered for defense in depth.

Security Design Considerations

(5) Security response.

The design of engineered physical security structures, systems, and components performing neutralization functions and engineered fighting positions relied on to protect security personnel performing neutralization functions should provide overlapping fields of fire. The design configuration should provide layers of opportunities for security response, with each layer assuring that a single failure does not result in the loss of capability to neutralize the design basis threat adversary.

Security Design Considerations

(6) Control measures protecting against land and waterborne vehicle bomb assaults.

The design of physical security structures, systems, and components, in conjunction with site-specific natural features, that are relied on to protect against a design basis threat land vehicle and waterborne vehicle bomb assault should provide assurance for the protection of the reactor building and structures containing safety related structures, systems, and components from explosive effects that are based on the maximum design basis threat quantity of explosives. The vehicle control measures (passive and active barrier systems) to deny land or waterborne vehicle bomb assaults should be located at a bounding minimum safe stand-off distance to adequately protect all structures, systems, and components required for safety and security.

Security Design Considerations

(7) Access control portals:

The design of access control portals should provide assurance of detecting and denying unauthorized access to persons and pass-through of contraband materials (e.g., weapons, incendiaries, explosives). The design should apply the principles of redundancy and diversity to achieve the intended control functions.

(8) Defense model architecture.

The design of the defensive architecture for digital systems and networks to protect against a cyber attack should establish the logical and physical boundaries between digital assets with similar risks and digital assets with lower security risks. Digital assets associated with safety, important to safety, and security functions should be located at the highest security level and protected from all lower levels.

Security Design Considerations

(9) Cyber security defense-in-depth.

A defense-in-depth protective strategy consisting of complementary and redundant cyber security controls should be employed to establish layers of protections to safeguard critical digital assets, critical systems, or both. The failure of a single protective strategy or security control should not result in the compromise of a safety, important-to-safety, security, or emergency preparedness function.

(10) Least functionality.

The design of the digital assets and digital communication systems should incorporate the principle of least functionality. The design should:

(1) Eliminate unused/unnecessary functionality, protocols, ports, and services capable of being used in a stage of a cyber attack; or

(2) Disable unused/unnecessary functionality, protocols, ports, and services and provide protections against enabling and use of the capabilities in a stage of a cyber attack; or

(3) Provide protections to prevent the use of unused/unnecessary functionality, protocols, ports, and services in a stage of a cyber attack when eliminating or disabling the capabilities is not practical.

Licensing Project Plans

- Initial familiarization interactions
- Coupled licensing and review plans
  - Priorities for critical decisions
  - Resource and schedule constraints
  - Routine interactions, monitor, and adjust
- Developing important reference documents (e.g., topical reports, codes and standards)
- Research plans
  (e.g., test reactors, qualification testing)

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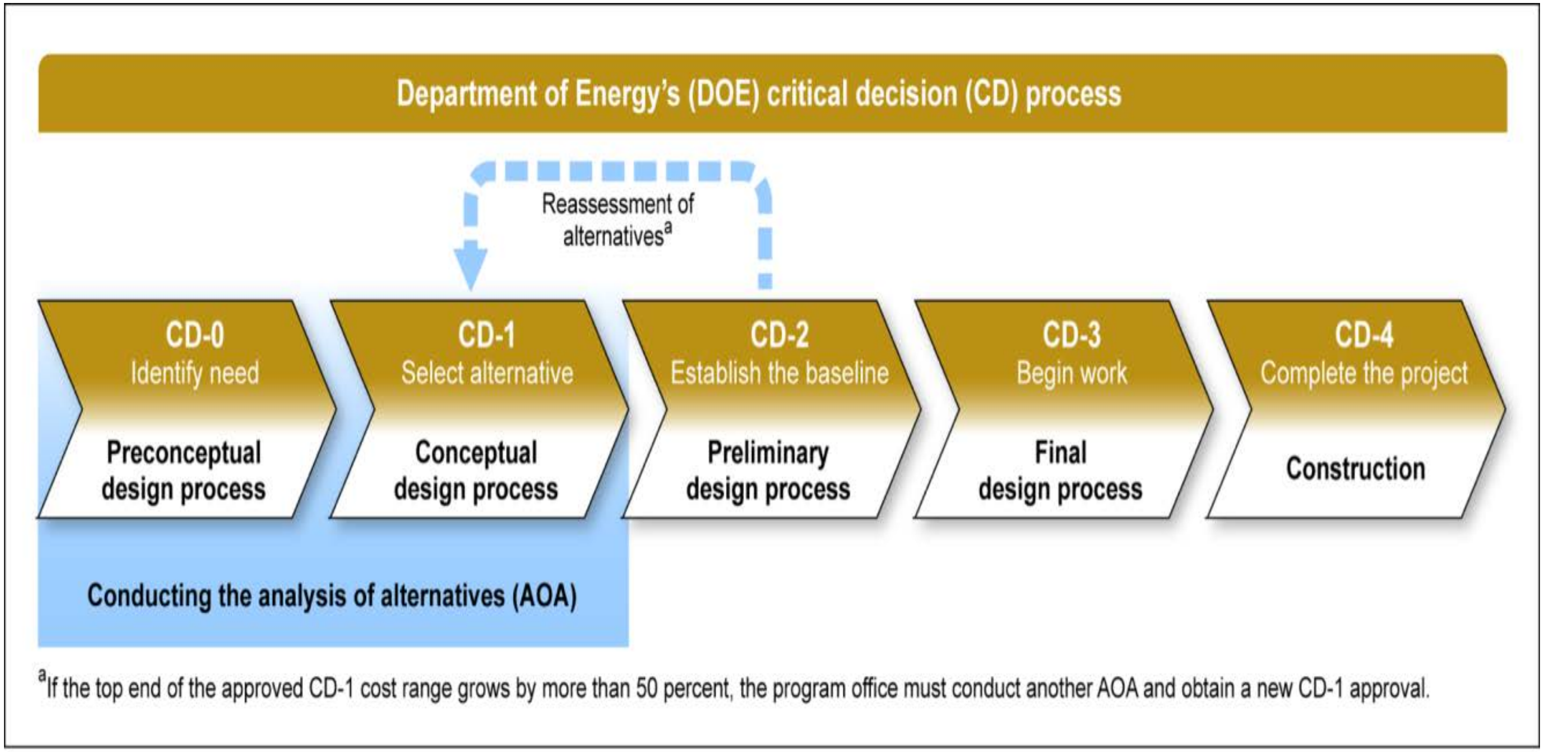
Strategy 3
(Activity 4: Flexible Approach, Roadmap)

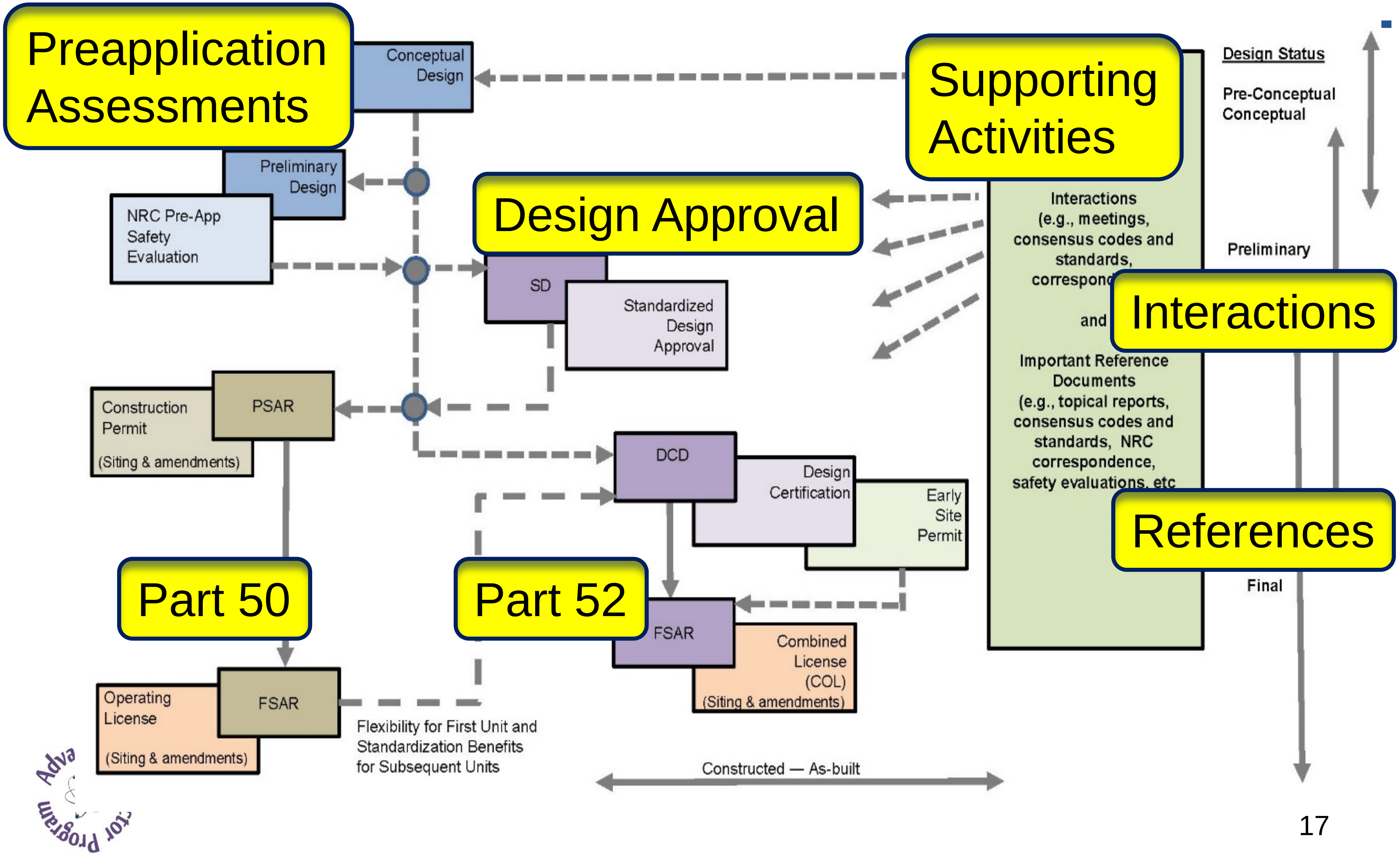
DOE Critical Decision Process



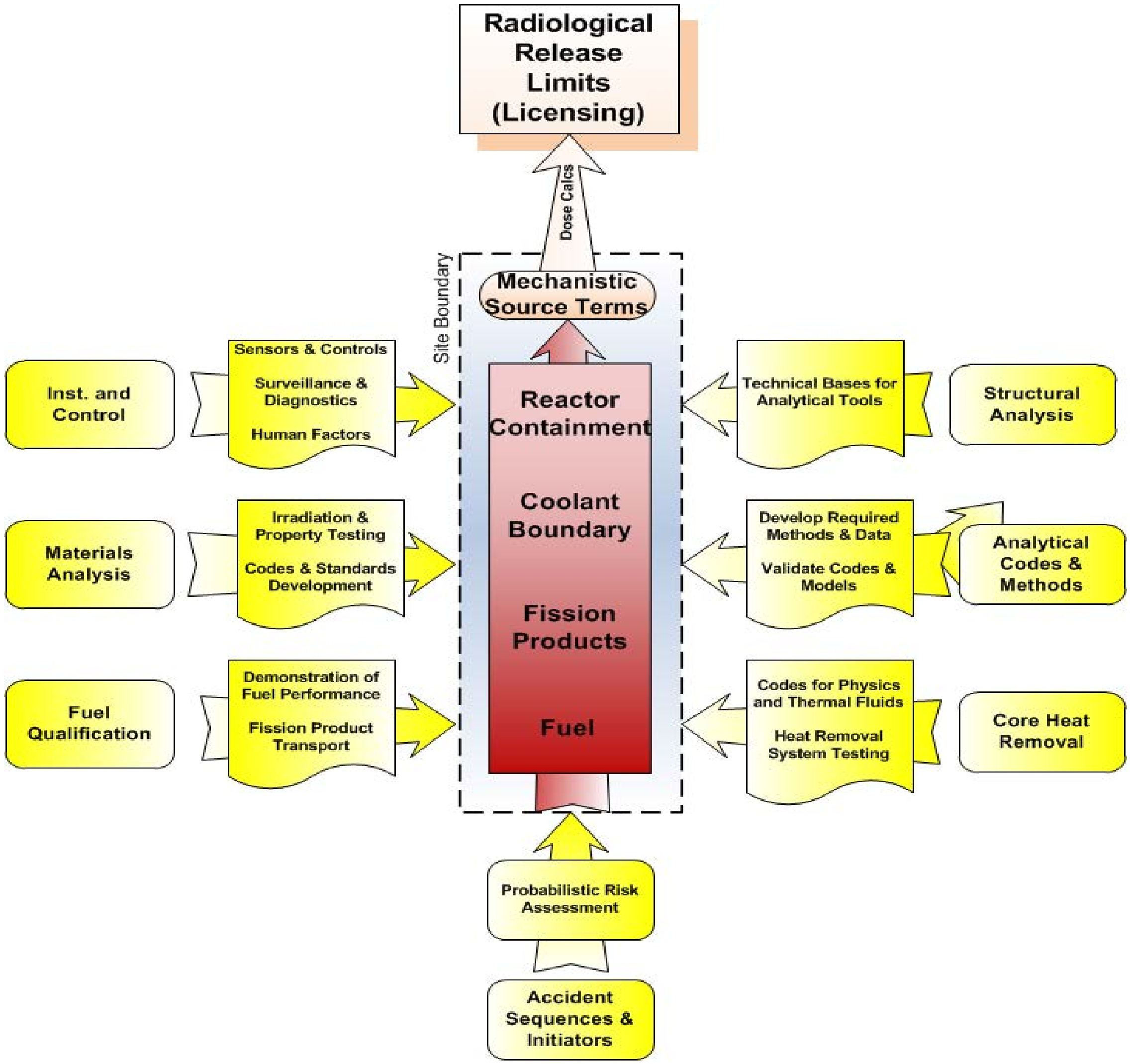
Source: GAO analysis of DOE's Order 413.3B. | GAO-15-37

Advanced Reactor Program

Flexible Approaches / Roadmap



Key Inputs for Licensing (INL Figure)



Preliminary (preapplication) Design Assessments

All or selected topics to support critical decisions

- General Description of the Plant
- Site Characteristics
- Design of SSCs and Equipment
- Reactor
- Reactor Coolant and Connecting Systems
- Engineered Safety Features
- Instrumentation and Controls
- Electric Power
- Auxiliary Systems
- Steam and Power Conversion System
- Radioactive Waste Management
- Radiation Protection
- Conduct of Operations
- Verification Programs
- Transient and Accident Analyses
- Technical Specifications
- Quality Assurance and Reliability Assurance
- Human Factors Engineering
- Probabilistic Risk Assessment/Severe Accident Evaluation

RG 1.206 Chapters 1-19

- Emergency Planning
- Security
- Staffing
- Mitigating Strategies
- Aircraft Impact Assessment
- Environmental Report
- Financial
- Inspections, Tests, Analyses, and Acceptance Criteria
- Insurance
- Fuel Cycle
- Other (design or technology specific)

Other Parts of Applications & Possible Issues

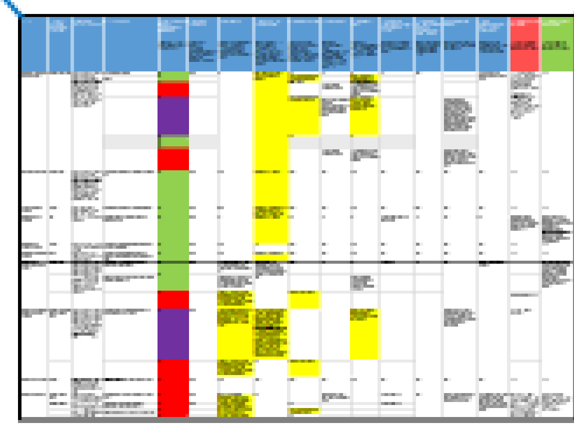
Organization/translation of design information into licensing basis information

Enhanced Safety Focused Review for SMRs

Key Review Considerations

- Safety-significance
- Regulatory compliance
- Novel design
- Shared structures, systems, and components
- Licensing approach
- Safety margin
- Defense-in-depth
- Operational programs
- Impact on safety functions
- Additional risk insights
- Other considerations

Review Tool



Output:
Scope and Depth of Review

- Provide supplemental approaches for implementation of NUREG-0800, Introduction - Part 2 and Design Specific Review Standard reviews
- Systematic thought process applicable to non-structure, system, or component and programmatic reviews

Department of Energy – Questions/Discussion

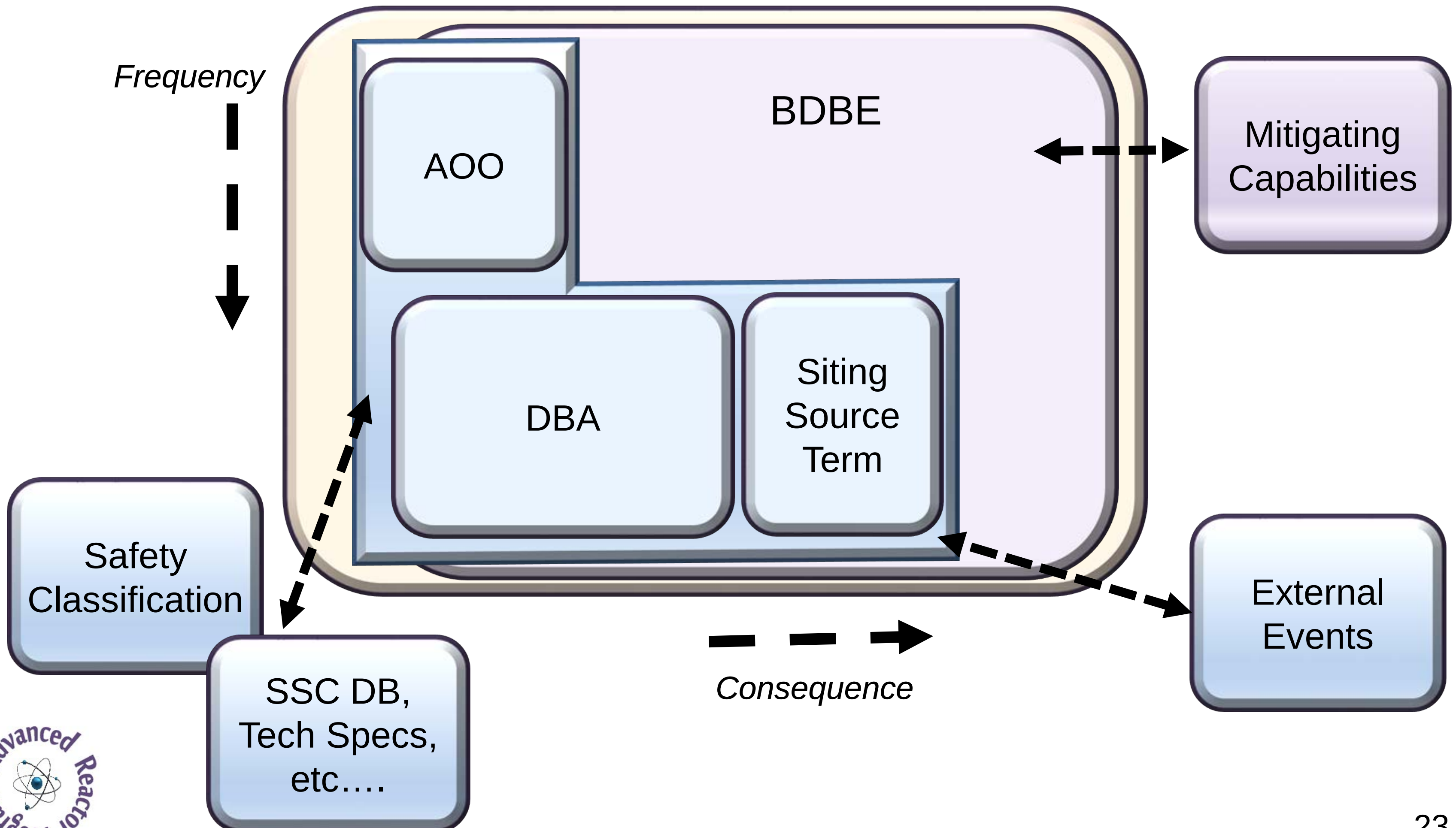
Licensing Basis Events

- ACRS Interactions
- Terminology
  - Deterministic
    - Selection
    - Analyses
  - Barriers (functional, physical)
- White Paper Relationships
  - Licensing Basis Events
  - Probabilistic Risk Assessment
  - Safety Classification
  - Defense in Depth

Strategy 3

(Activity 2: non-LWR licensing basis)

Current LBE Construct (LWRs)

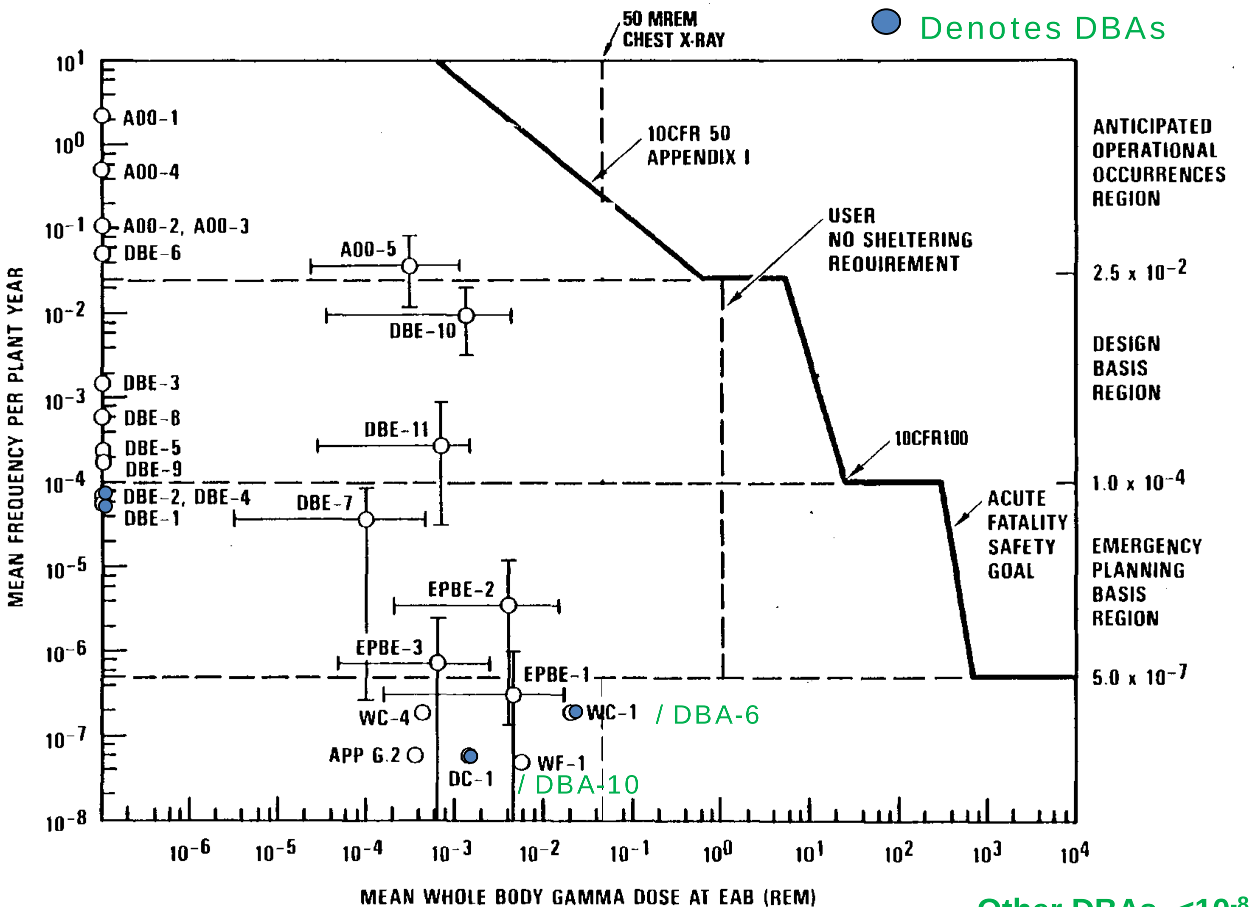


Strategy 3

(Activity 2: non-LWR licensing basis)

Example MHTGR LBEs, DBAs on F-C Plot (circa 1987)

(From 12/15/2016 NEI Presentation)



Other DBAs $<10^{-8}$

Licensing Basis Events Other Considerations

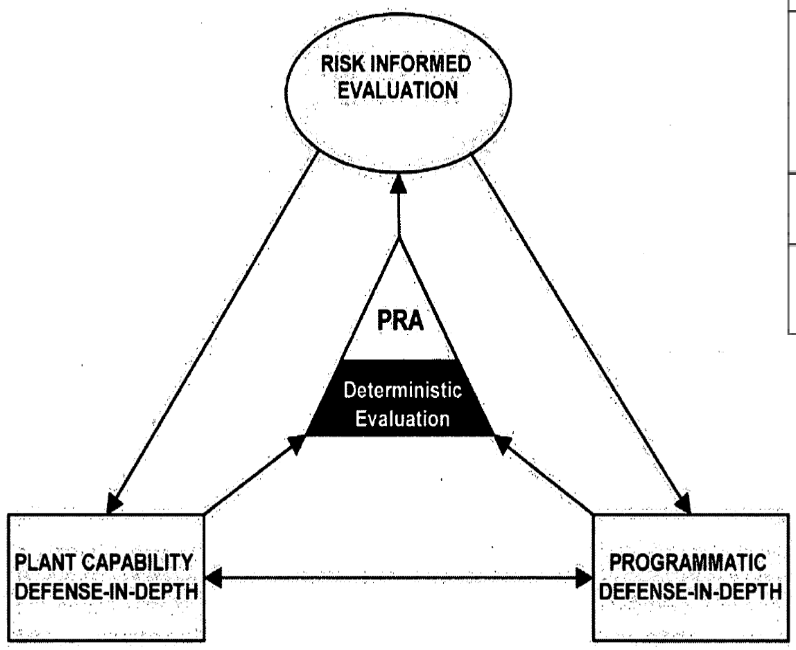


Figure E-1. Illustration showing the three major elements of the NGNP framework.

INL/EXT-09-17139
Next Generation Nuclear Plant
Defense-in-Depth Approach

Table 1-1
Safety Criteria and Analysis Requirements

Event	Region	No Fuel Melting	Fuel Cladding Integrity	Core Coolable Geometry	Primary Coolant Boundary Integrity	Containment Integrity	Radiation Dose at EAB and LPZ	Analysis Requirements
DBE	AOO	✓	✓	*	✓ ASME SL† "B," "C"	*	*	Conservative
	DBA	-	✓	✓*	✓ ASME SL "D"	✓	*	Conservative
Beyond DBE	ATWS	-	✓	✓*	✓ ASME SL "D"	✓	✓*	Best estimate plus uncertainties

Notes:
✓ Explicit safety criterion is defined.
* Meeting the safety criterion is expected if the previous criterion is met.
† SL = Service Level

From "4S Safety Analysis"
submitted by Toshiba Corporation, July 28, 2009
ADAMS Accession No. ML092170507

Advanced Reactors Developers & NRC Policy Issues

Jeffrey S. Merrifield
Chairman, Advanced Reactors Task Force
U.S. Nuclear Infrastructure Council

NRC Advanced Reactors Public Meeting
March 22, 2017

Overview

- NIC commends the staff of the U.S. Nuclear Regulatory Commission (NRC) for organizing this meeting
- We appreciate the continuing opportunity to share our views on these important issues
- NIC serves as a leading advocate for Advanced Reactor technologies
- We appreciate the efforts the NRC staff has made on this regulatory process
- NIC's comments today will focus on the "policy issues" facing Advanced Reactors as identified at the March 8, 2017 NRC ACRS Future Plant Designs Subcommittee Meeting (Non-LWR Vision and Strategy Implementation Action Plans) and our related thoughts about the NRC Advanced Reactor licensing process
- The comments we will share today reflect a survey conducted of NIC's Technology Owners Group – representing the views of 16 Advanced Reactor technology developers

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Policy Issues

License for Prototype Reactors	Drafting white paper
License Structure for Multi-Module Facilities	SECY-11-0079
Appropriate Source Term, Dose Calculations, and Siting	SECY-16-0012
Offsite Emergency Planning (EP) Requirements	SECY-15-0077 Drafting Regulatory Basis
Annual Fees	Final Rule (May 2016)
Insurance and Liability	Evaluating for periodic report to Congress on Price-Anderson Act
Manufacturing License Requirements	SECY-14-0095 (SMRs)
Use of Probabilistic Risk Assessment in the Licensing Process	SRP Revisions (safety focused review)

Policy Issues

Key Component and System Design Issues	Design Specific
Operator Staffing for Small or Multi-Module Facilities	SECY-11-0098 (flexibility w/ existing guidance)
Operational Programs for Small or Multi-Module Facilities	SECY-11-0112 (flexibility w/ existing guidance)
Installation of Reactor Modules During Operation of Multi-Module Facilities	SECY-11-0112 (existing guidance)
Industrial Facilities Using Nuclear-Generated Process Heat	SECY-11-0112 (assess as necessary)
Decommissioning Funding Assurance	SECY-11-0181 (Site-specific exemptions)
Implementation of Defense-In-Depth (DiD) Philosophy for Advanced Reactors	SECY-15-0168 (part of licensing framework)

Policy Issues

Security and Safeguards Requirements for SMRs	Ongoing discussions (NEI White Paper)
Aircraft Impact Assessments	Ongoing discussions
Licensing Basis Event Selection	Ongoing discussions
Functional Containment Performance Criteria	Ongoing discussions
Fuel qualification, materials qualification	Issues vary by technology
Fuel cycle facilities, enrichments	Ongoing discussions
Continuing efforts to identify and prioritize technical and policy issues	

Additional Policy Issues

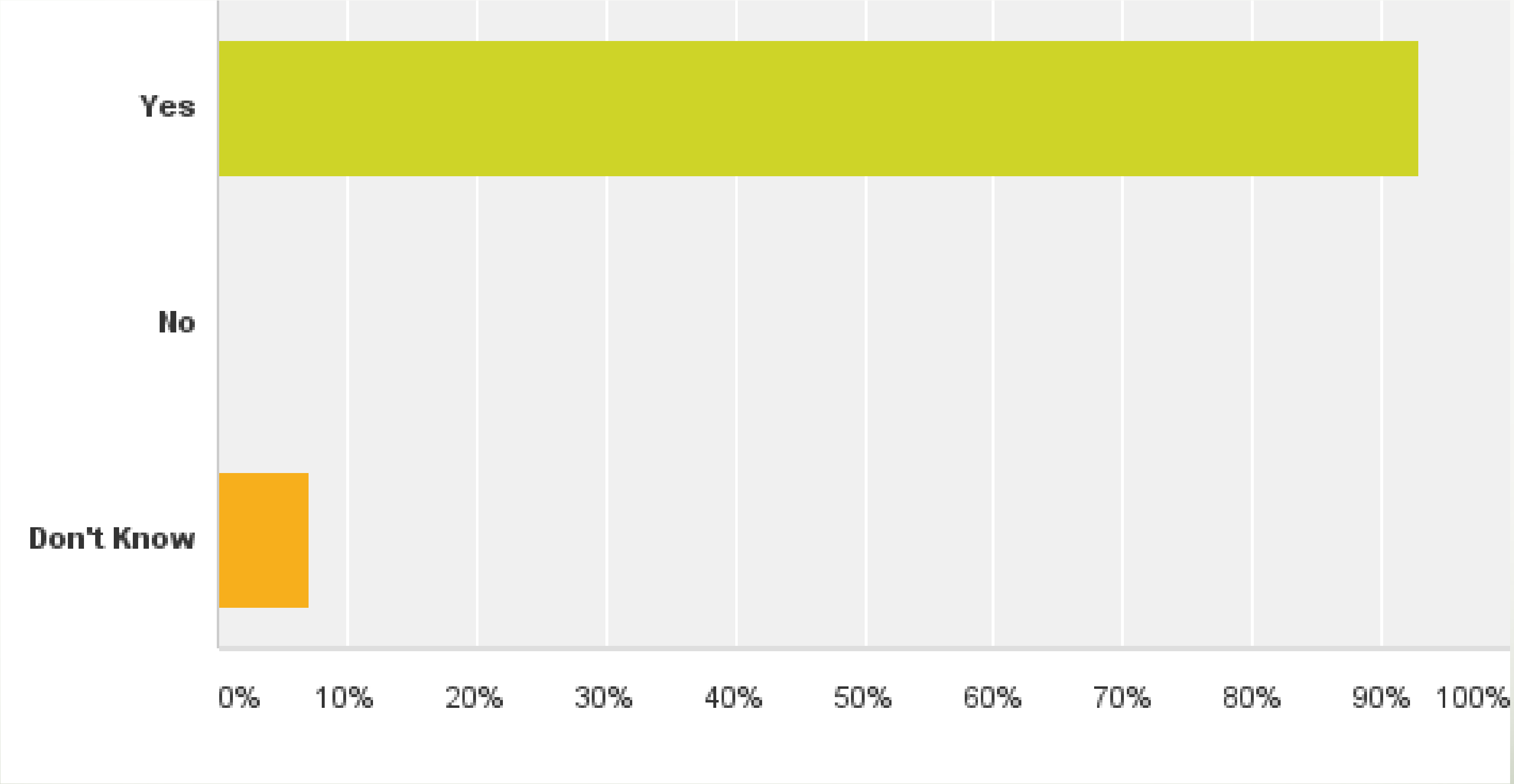
(Not identified in the NRC 3/8/17 presentation)

Used of High-Assay Low Enriched Uranium	This issue is currently being evaluated by the NRC staff and will likely be a matter of increasing dialog over the coming months.
Advanced reactor licensing technical support (engineering, design, and documentation to support licensing)	
Life-of-plant on-site spent fuel storage/disposition	

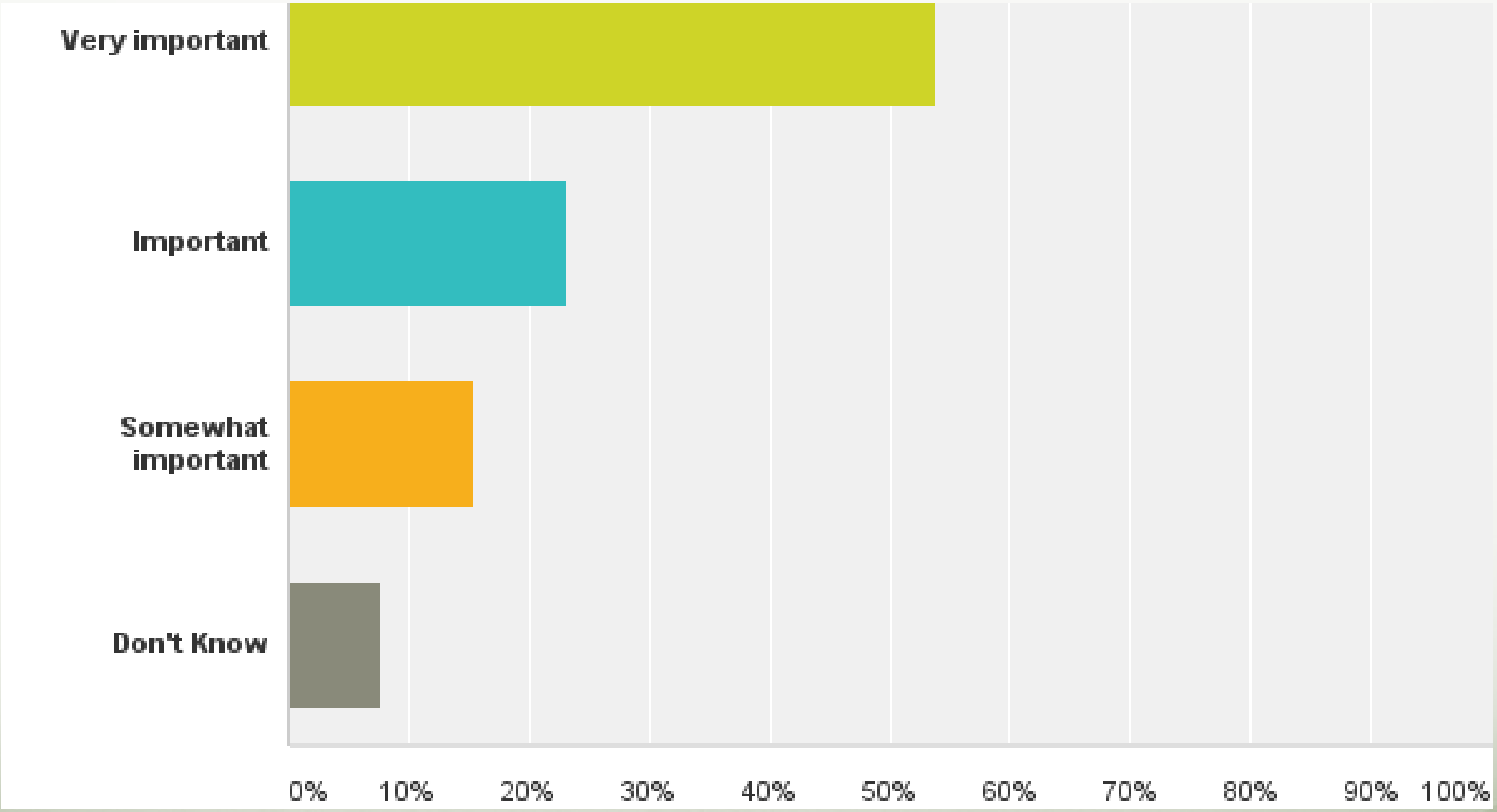
USNIC AR Developer Survey

- USNIC conducted a survey of 16 leading U.S. Advanced Reactor technology developers with regard to relevant NRC Advanced Reactor policy issues.
- This was a blind survey so individual results were not identified
- 88% of the developers surveyed provided input

Q1: Is NRC resolution of outstanding generic advanced reactors licensing "policy issues" important to your company?



Q2: How important is NRC resolution of outstanding generic advanced reactors licensing "policy issues" to your company?



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Q3: Of the 24 Policy Issues listed below, please rank their individual importance?

	High importance	Important	Low importance	Not sure	Total	Weighted Average
Appropriate Source Term, Dose Calculations, and Siting	**64.29%** 9	**28.57%** 4	**7.14%** 1	**0.00%** 0	14	1.43
Use of Probabilistic Risk Assessment in the Licensing Process	**61.54%** 8	**23.08%** 3	**15.38%** 2	**0.00%** 0	13	1.54
Licensing Basis Event Selection	**46.15%** 6	**46.15%** 6	**7.69%** 1	**0.00%** 0	13	1.62
Fuel cycle facilities, enrichments	**57.14%** 8	**21.43%** 3	**21.43%** 3	**0.00%** 0	14	1.64
Functional Containment Performance Criteria	**53.85%** 7	**23.08%** 3	**23.08%** 3	**0.00%** 0	13	1.69
Fuel qualification, materials qualification	**42.86%** 6	**42.86%** 6	**14.29%** 2	**0.00%** 0	14	1.71

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Q3: Of the 24 Policy Issues… continued

	High importance	Important	Low importance	Not sure	Total	Weighted Average
Advanced Reactor Licensing Technical Support (engineering, design, and documentation to support licensing) - (note: this issue is not included in the NRC list of "policy issues")	**38.46%** 5	**46.15%** 6	**15.38%** 2	**0.00%** 0	13	1.77
Key Component and System Design Issues	**41.67%** 5	**33.33%** 4	**25.00%** 3	**0.00%** 0	12	1.83
Operator Staffing for Small or Multi-Module Facilities	**46.15%** 6	**23.08%** 3	**30.77%** 4	**0.00%** 0	13	1.85
Industrial Facilities Using Nuclear-Generated Process Heat	**35.71%** 5	**42.86%** 6	**21.43%** 3	**0.00%** 0	14	1.86
Operational Programs for Small or Multi-Module Facilities	**46.15%** 6	**15.38%** 2	**38.46%** 5	**0.00%** 0	13	1.92
Offsite Emergency Planning (EP) Requirements	**23.08%** 3	**53.85%** 7	**23.08%** 3	**0.00%** 0	13	2.00

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Q3: Of the 24 Policy Issues… continued

	High importance	Important	Low importance	Not sure	Total	Weighted Average
Annual Fees	**30.77%** 4	**38.46%** 5	**30.77%** 4	**0.00%** 0	13	2.00
Implementation of Defense-In-Depth (DiD) Philosophy for Advanced Reactors	**41.67%** 5	**25.00%** 3	**25.00%** 3	**8.33%** 1	12	2.00
Security and Safeguards Requirements for SMRs	**38.46%** 5	**23.08%** 3	**38.46%** 5	**0.00%** 0	13	2.00
Use of High-Assay Low Enriched Uranium (note: this issue is not included in the NRC list of "policy issues")	**30.77%** 4	**38.46%** 5	**30.77%** 4	**0.00%** 0	13	2.00
License for Prototype Reactors	**23.08%** 3	**38.46%** 5	**38.46%** 5	**0.00%** 0	13	2.15
License Structure for Multi-Module Facilities	**21.43%** 3	**35.71%** 5	**42.86%** 6	**0.00%** 0	14	2.21
Insurance and Liability	**23.08%** 3	**30.77%** 4	**46.15%** 6	**0.00%** 0	13	2.23

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Q3: Of the 24 Policy Issues… continued

	High importance	Important	Low importance	Not sure	Total	Weighted Average
Aircraft Impact Assessments	**15.38%** 2	**46.15%** 6	**38.46%** 5	**0.00%** 0	13	2.23
Installation of Reactor Modules During Operation of Multi-Module Facilities	**7.69%** 1	**30.77%** 4	**61.54%** 8	**0.00%** 0	13	2.54
Decommissioning Funding Assurance	**0.00%** 0	**46.15%** 6	**53.85%** 7	**0.00%** 0	13	2.54
Manufacturing License Requirements	**0.00%** 0	**30.77%** 4	**69.23%** 9	**0.00%** 0	13	2.69
Life-of-plant on-site spent fuel storage/disposition (note: this issue is not included in the NRC list of "policy issues")	**0.00%** 0	**30.77%** 4	**69.23%** 9	**0.00%** 0	13	2.69

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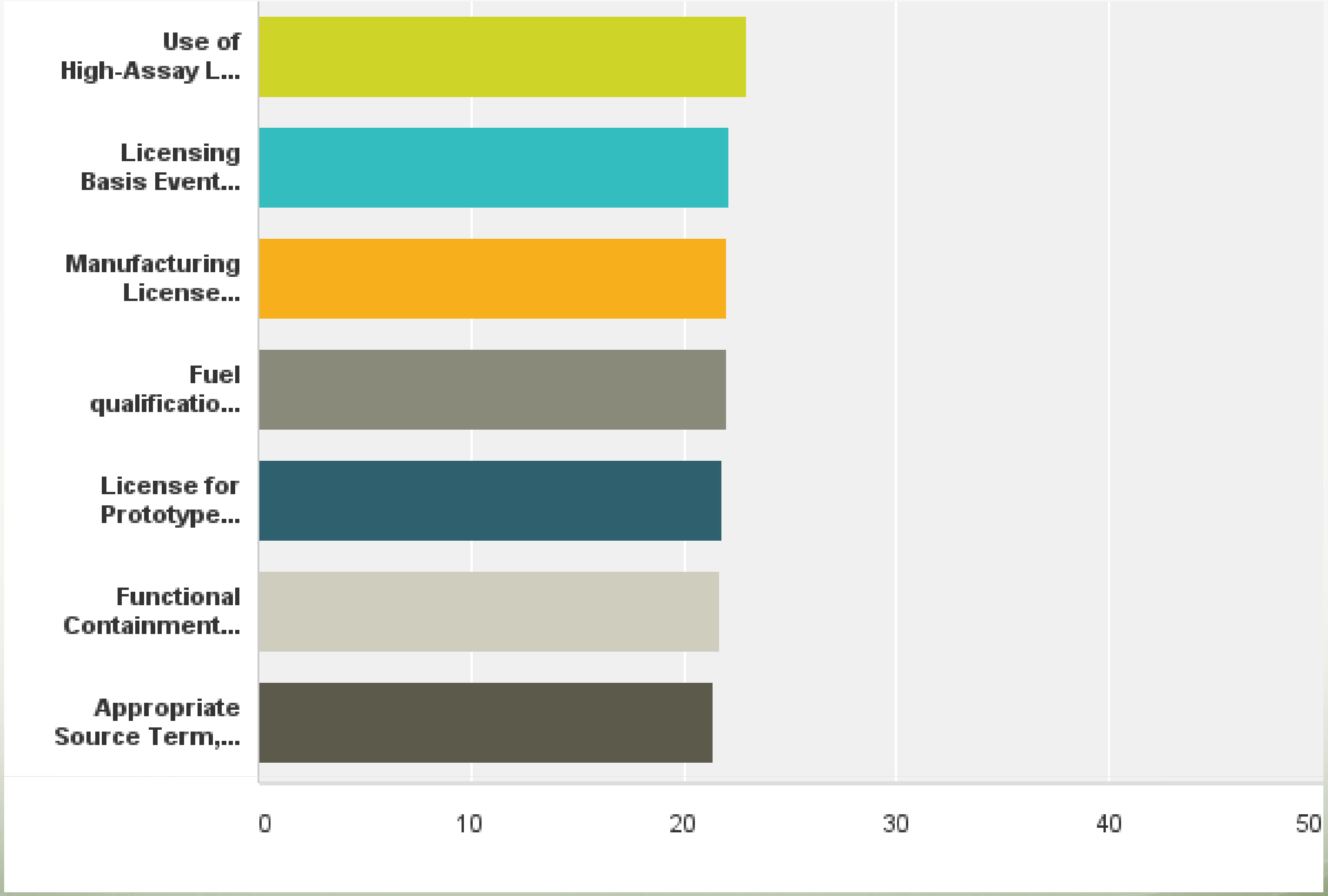
Q4: Please rank the five most important policy issues to your company



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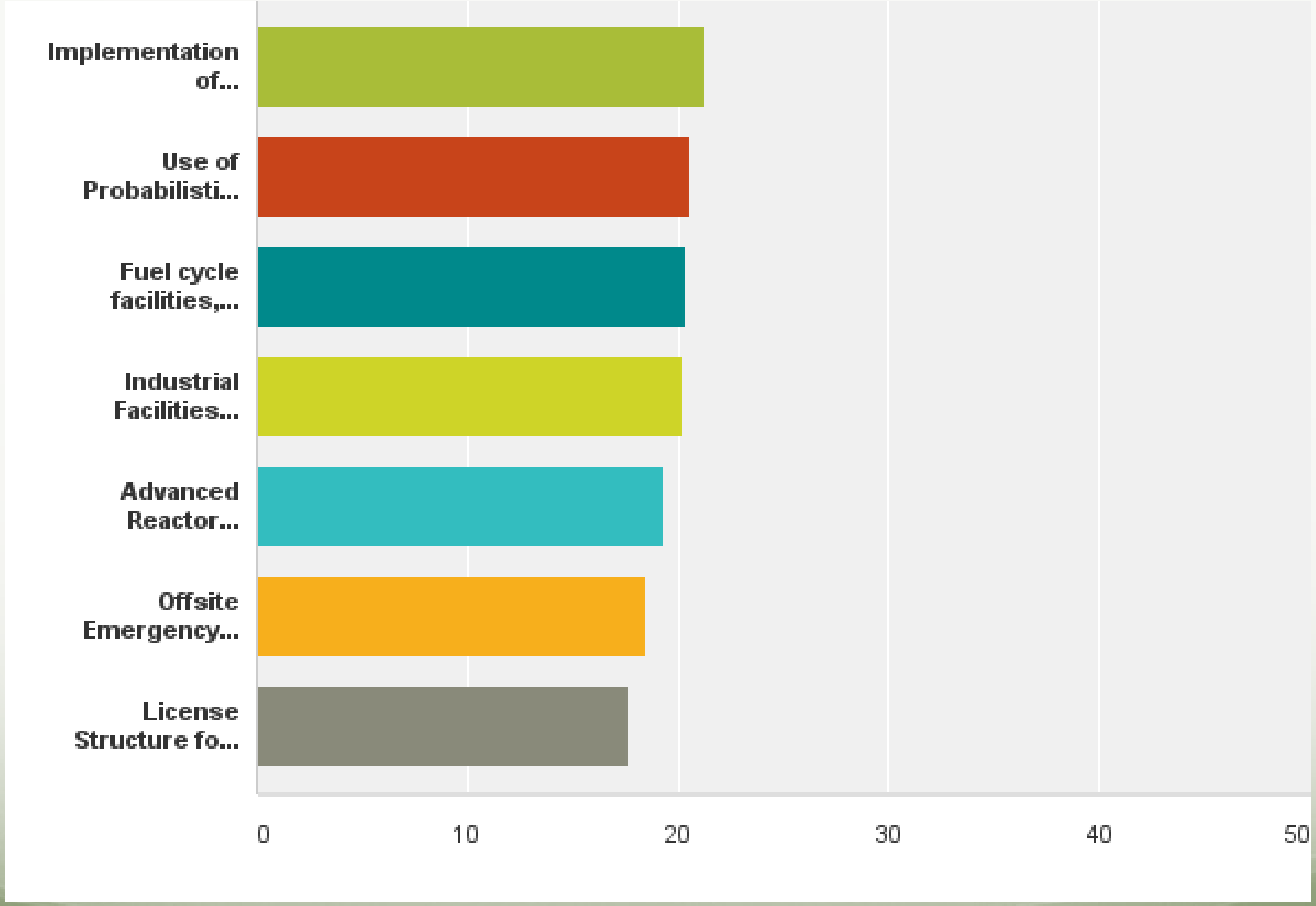
Q4: Please rank the five most important policy issues… continued



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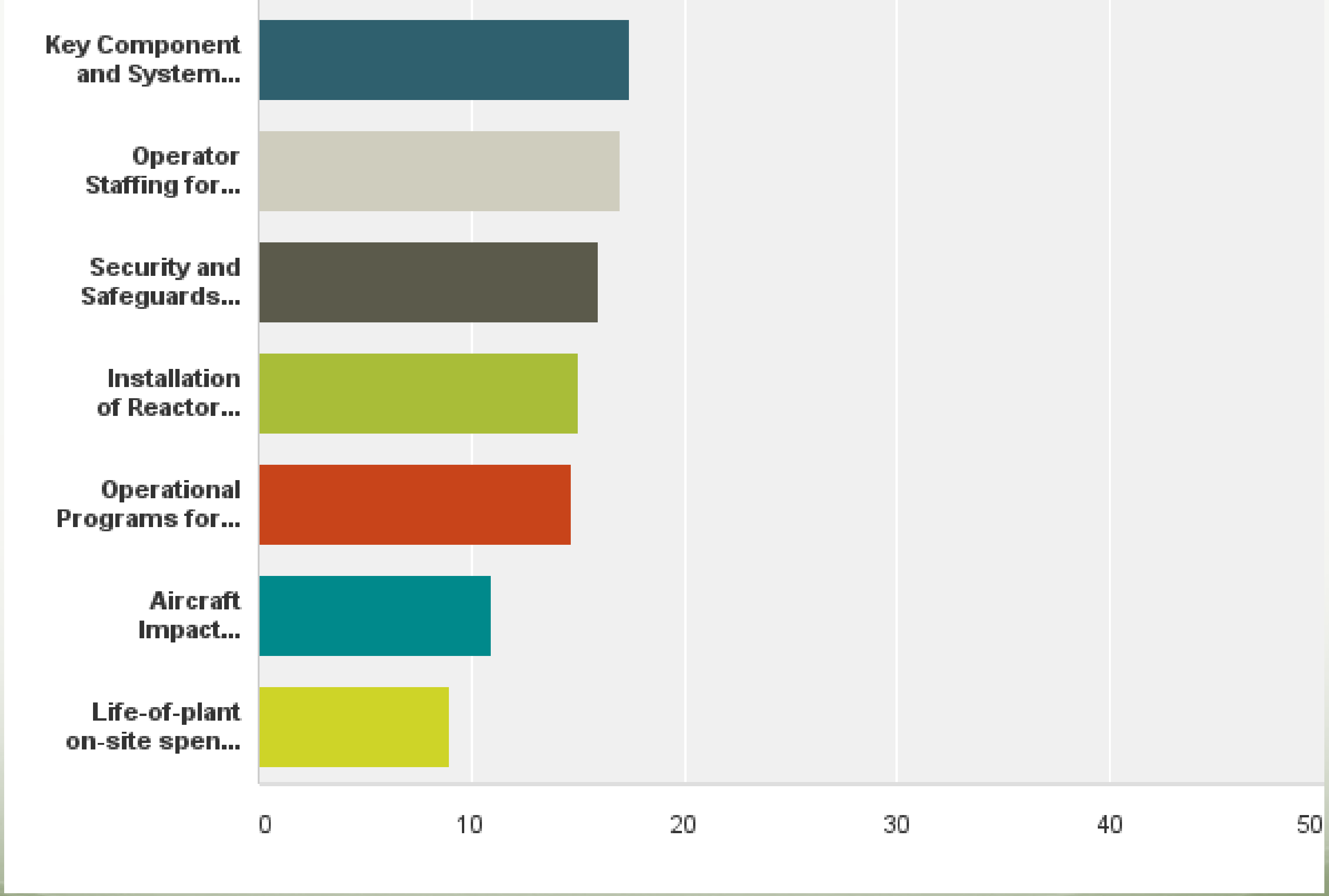
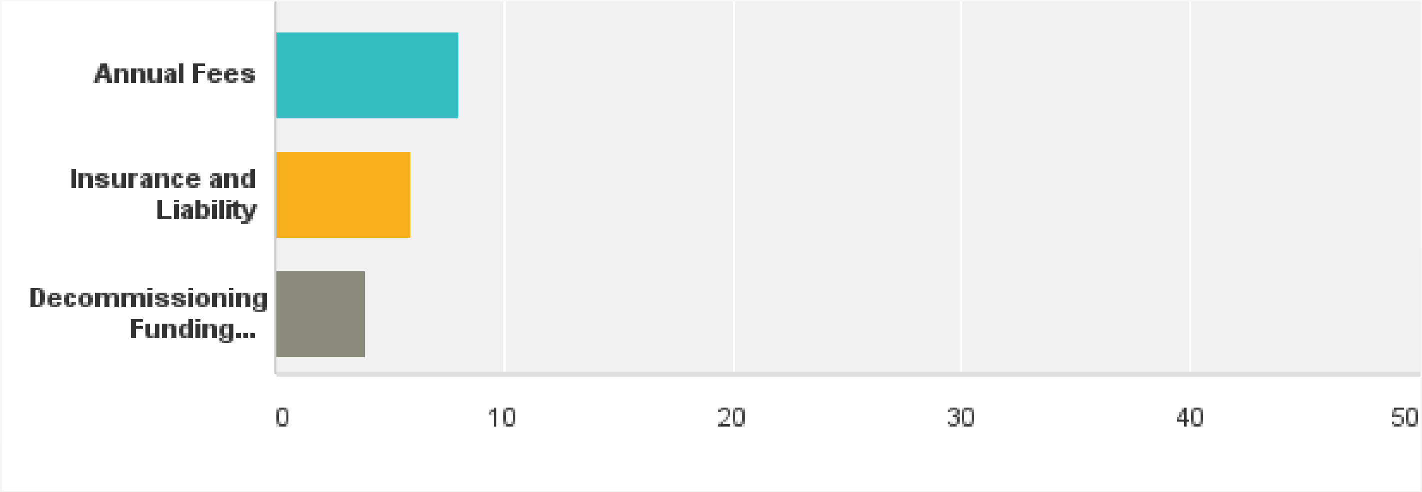
Q4: Please rank the five most important policy issues… continued



Q4: Please rank the five most important policy issues… continued



Q5: Are there any policy issues that are not currently included on this list?

- Concern about the length of ACRS Reviews – 6-12 months added time
- Breath of information required on the docket
- Use of Adequate Protection Standard
- Pre-Application policy meeting to address policy issues similar to what NRC did for SMRs
- Develop pre-licensing framework to allow discussion without "getting on the NRC clock"
- Need to address LEU issues near term
- Need MSR guidelines

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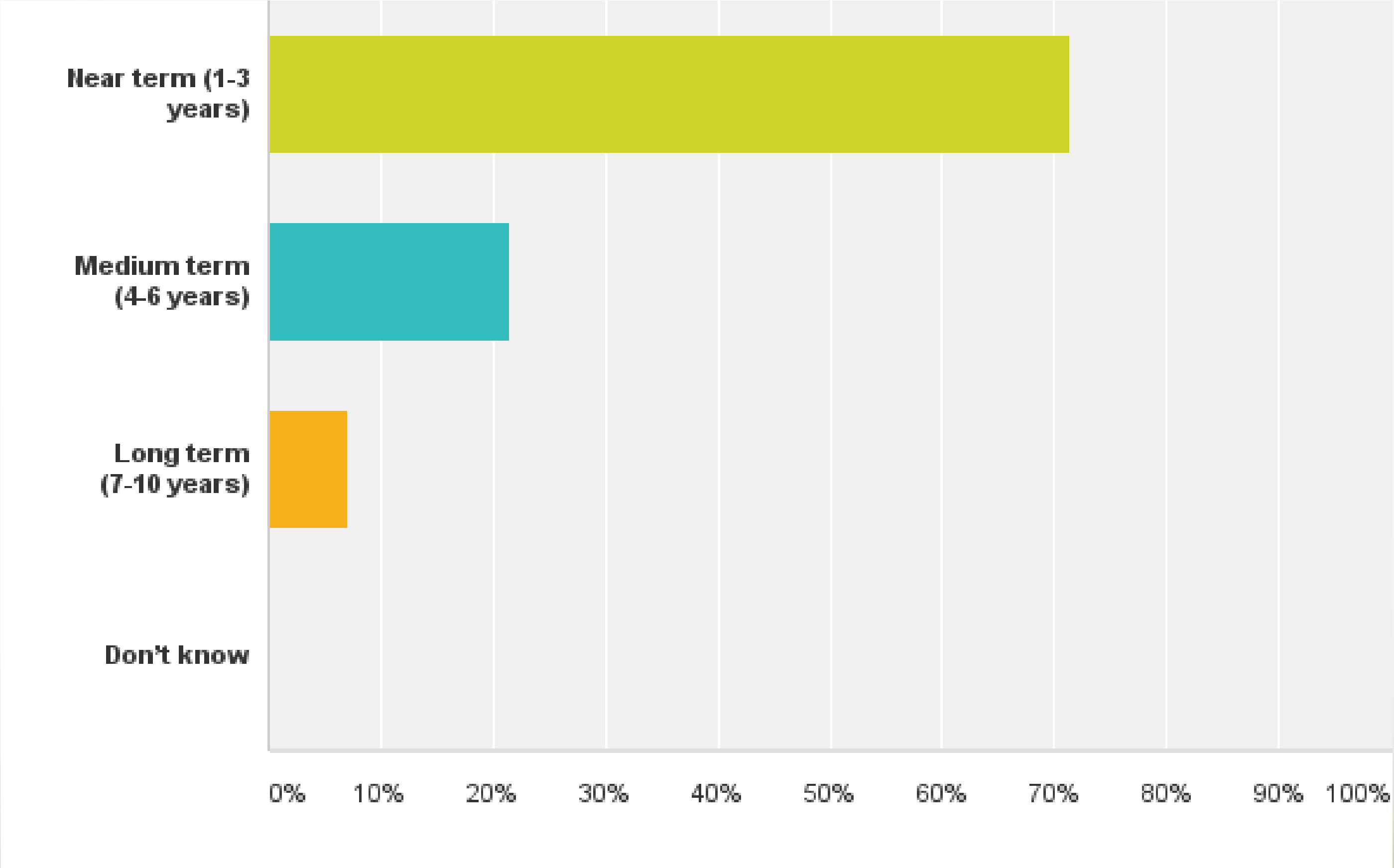
Q6: In what time frame do these policy issues need to be resolved by the government?



Q7: Actions that the NRC and/or industry can do to resolve these issues?

- Develop roadmap for high-assay LEU licensing, including transportation regulations
- Use lessons-learned from recent Part 52 reviews, license amendments to "re-establish" expectations for level-of-detail and structure for applications (too much information is captured in FSARs; need to focus on safety significant issues)
- Dedicate a full time PM and assign dedicated technical staff resources to each of these policy issues. Prepare individual "Issue Resolution" Implementation Plans.
- Get NRC to commit to specific timeline to comment and accept proposals

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Q7: Actions that the NRC and/or industry can do to resolve these issues?

- Government investment to train NRC on Advanced Reactor technology - avoid training through RAI's
- Need near-term Commission policy direction on Part 100 LPZ and population center distance requirements in order to support process heat applications
- Move towards risk informed performance based regulation

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About the USNIC

- Leading business consortium advocate for increased U.S. nuclear energy use and global deployment of U.S. nuclear technologies and services
- Represents nearly 90 member companies encompassing wide representation of the nuclear energy supply chain and key movers
- Member of the Civil Nuclear Trade Advisory Committee, the U.S. Industry Delegation to the IAEA and the ANS International Committee
- Strongly supports Gen 3+ reactors, small modular reactors and advanced reactors moving in parallel paths

United States NIC Nuclear Infrastructure Council

The United States Nuclear Infrastructure Council (USNIC) is the leading U.S. business consortium advocate for nuclear energy and promotion of the American supply chain globally. Composed of nearly 90 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 seven working groups and select task forces. For more information visit www.usnic.org

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Update on “Major Portions” Project (Standard Design Approval)

NRC Meeting on Regulatory Process Improvements for Advanced Reactor Designs

March 22, 2017

Background

- 10 CFR Part 52, Subpart E allows an applicant to seek standard design approval for either an entire plant or "major portions" thereof
- NIA supporting NRC in clarifying the meaning of "major portions" to make SDA process useful for advanced reactor developers
- Drafting products for review, revision, etc., with input from industry representatives (and NEI/ARWG) for delivery to the NRC for their initial review

NUCLEAR INNOVATION ALLIANCE

Schedule/Milestones

- Phase 1 (Mar-Apr 2017) provides summary white paper describing SDA process at high level in support of NRC strategy/roadmap documents (in review)
- Phase 2 (proposed ~Apr-May 2017): expands into more detail to provide additional guidance for use in LPP/REP
- Phase 3 (proposed - TBD): expands into ISG or other NRC guidance, including detailed discussion of boundary conditions and integration with Southern-led regulatory framework initiative

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Phase 1 Report Topics

- Purpose/benefit of SDA
- Scope
- Criteria for selection of “Major Portions”
- Defining interfacing system boundary conditions
- Context within “staged licensing”
- Regulatory basis & precedent
- Practicality considerations
- Risks and mitigation
- Regulatory analogs

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Issues for Discussion

- Use of "preliminary design"
  - Consistent with NRC roadmap
  - "Standard design" – final vs. preliminary
  - Developer/applicant bears risk relative to future changes (true for "final" or "preliminary")
- Scenario options (by example)
  - "Typical" example: broad set of functional safety requirements
  - More limited "portion" of safety basis with corresponding limited SER (e.g., for SSCs with high regulatory risk)
  - "Major portion" for site parameter, e.g., GMRS, structural design
  - "Major portions" for other than safety-related SSCs, e.g., molten salt processing systems, waste or new fuel processing, or security

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Thank you

Feedback & Questions

Please feel welcome to send additional input at any time to Ashley Finan (afinan@catf.us).

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Utility-Led Initiative for Modernization of Technical Requirements for Licensing of Non-Light Water Reactors

Amir Afzali

March 22, 2017 • USNRC Rockville MD

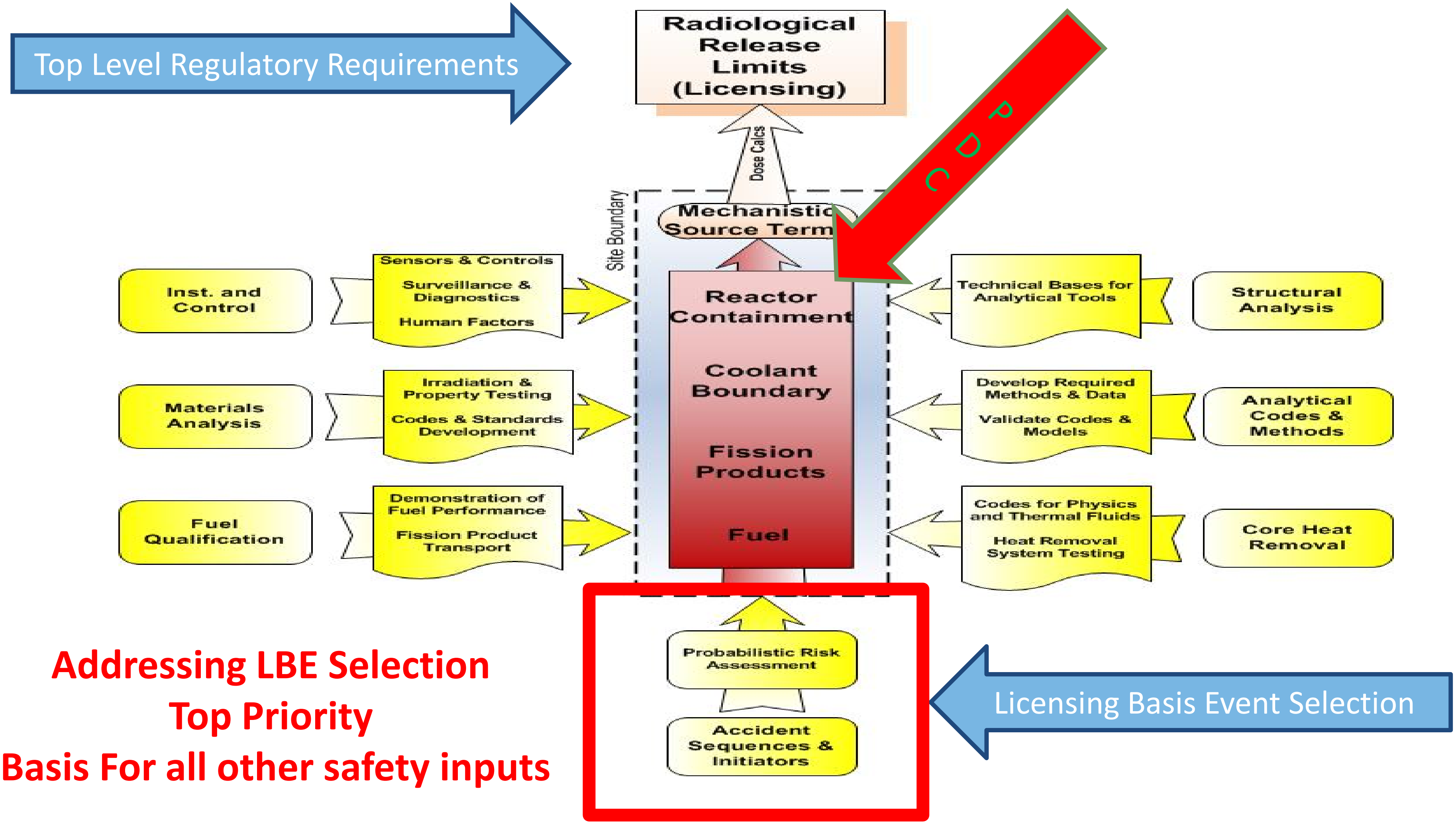
Overview of Session

- LMP Background for LBE Selection
- LBE Selection Modernization Objectives
- LBE Selection Process and related NRC Questions
- Future NRC Question Discussions

LMP LBE Selection Objectives

- Objective 1- Provide a Systematic Process that Meets Principles of Good Regulation More Clearly and Enhances Overall Safety
  - This objective is met by proposing to replace the current ad hoc RIPB process by a systematic RIPB process.
- Objective 2- Provide a Stable Process for Designs to be Developed from Concept to the Final Design
- Objective 3- Provides a Reliable Process for Balancing Safety Improvements and Burden Reduction During the Original Design and Licensing as well as when Unknowns Become Knowns
  - This focus is believed to be met through systematic RIPB SSC classification and DiD consideration as well as LBE Selection.

Objectives 1 & 2: Key Licensing Inputs



Addressing LBE Selection
Top Priority
Basis For all other safety inputs

Objective 1- The Key Consideration

- SRP Chapter 15.0 statement:

  *"If **the risk of an event** is defined as the product of **the event's frequency of occurrence and its consequences**, then the design of the plant should be such that all the AOOs and postulated accidents produce about the same level of risk (i.e., the risk is approximately constant across the spectrum of AOOs and postulated accidents). This is reflected in the **general design criteria (GDC), which generally prohibit relatively frequent <u>events</u> (AOOs) from resulting in serious consequences, but allow the relatively <u>rare events</u> (postulated accidents) to produce more severe consequences**."*

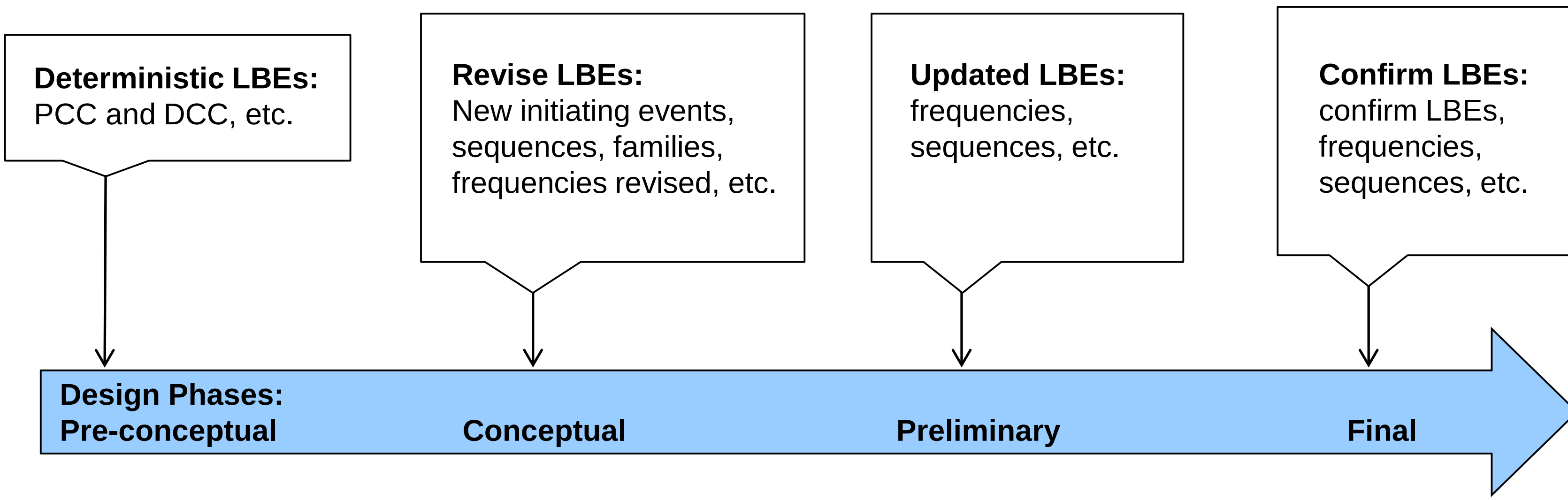
- Conclusion: To meet this requirement LBE Selection has to be RIPB
- Options: Ad hoc engineering judgement Approach vs. Systematic RIPB Process

Objective 1- Comparison of Options for the LBE Selection Process

LBE Selection Options	Process	Tools used for identification and consequence analysis	Frequency estimate	Uncertainty Analysis/Defense-in-depth	Quality Control
Ad Hoc RIPB	Events are identified and analyzed based on Engineering Judgment	FMEA, PIRT	Engineering judgment	Not explicitly identified, limited treatment, if any, based on engineering judgment.	Difficult to define
Systematic RIPB	Events are identified and analyzed based on Engineering Judgment and holistic system logic, including common cause Failures	FMEA, PIRT, PRA methods	Engineering judgment and data analysis	Explicitly identified and listed. Systematically analyzed and accounted for	ASME non-LWR PRA standard, EPRI work

Focus 2- Event Selection Timeline

LBE evolution by design phase:



LBE selection process inputs vary by design phase:

- Initial design concept*****
- Prior HTGR experience and PRAs*****
- Expert insights*****

- Basic design*****
- Initial analyses (FMEA, HAZOPs, etc)*****
- Initiate PRA development†
- Design rqmts.*****
- Expert reviews*****

- Updated design*****
- Detailed FMEAs, etc.*****
- Preliminary PRA results†
- Expert reviews*****
- Regulator interaction*****

- Mature design
- Detailed FMEAs, etc.
- Complete PRA results
- Expanded PRA scope†
- Expert reviews
- Regulator feedback

*** Steps performed during MHTGR project through early preliminary design**
† PRA scope and level of detail expands as design matures

Next Generation Nuclear Plant

Southern-Led Initiative for Modernization of Technical Requirements for Licensing of Non-Light Water Reactors

Elements of LBE Selection Approach Relevant to NRC Questions

Karl Fleming

Regulatory Process Improvements for Advanced Reactor Designs

March 2017 • USNRC Rockville MD

LBE Selection Attributes

- Systematic and reproducible process
- Sufficiently complete set of LBEs
- Timely input to design decisions
- Risk-informed and performance-based
- Reactor technology inclusive
  - Capable of identifying reactor specific safety issues
  - Applicability to wide range of non-LWR concepts
  - Uniform level of safety
- Capable of meeting applicable regulatory requirements

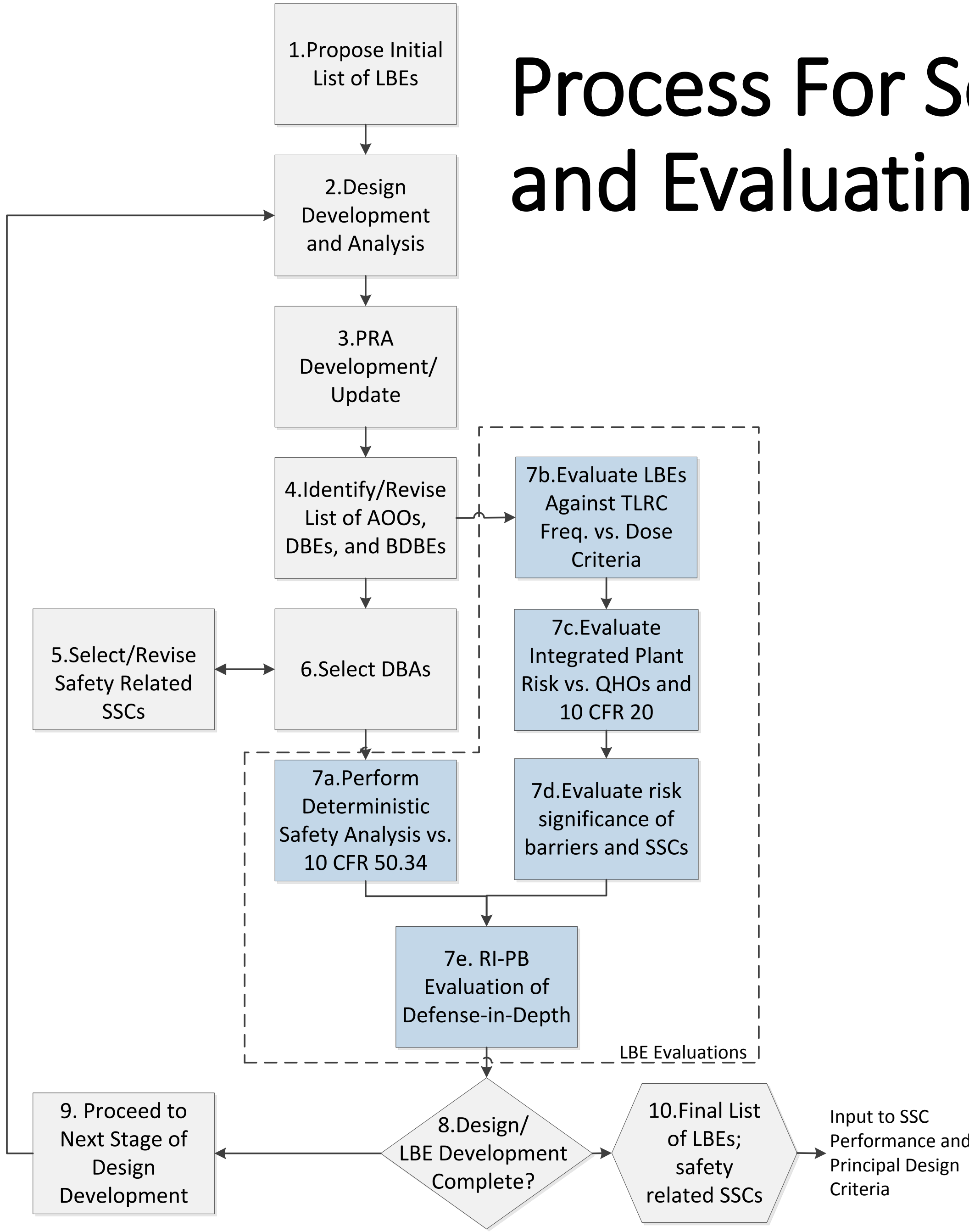
LBE Category Definitions

Event Type	NRC Definition	LMP Definition
Anticipated Operational Occurrences (AOOs)	"Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." [SRP 15.0 and 10 CFR 50 Appendix A]	Conditions of plant operation, events, and event sequences that are expected to occur one or more times during the life of the nuclear power plant which may include one or more reactor modules. Events and event sequences with frequencies of $1x10^{-2}$ per plant year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant regardless of safety classification.
Design Basis Events (DBEs)	"Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures." [SRP 15.0]	Events and event sequences that are expected to occur one or more times in the life of an entire fleet of nuclear power plants, but less likely than an AOO. Events and event sequences with frequencies of $1x10^{-4}$ per plant year to $1x10^{-2}$ per plant year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification. The objective and scope of the DBEs to form the design basis of the plant is the same as in the NRC definition. However DBEs do not include normal operation and AOOs as defined in the NRC references.
Beyond Design Basis Event (BDBE)	"This term is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. (In that sense, they are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand.) As the regulatory process strives to be as thorough as possible, "beyond design-basis" accident sequences are analyzed to fully understand the capability of a design." [NRC Glossary]	Events and event sequences that are not expected to occur in the life of an entire fleet of nuclear power plants. Events and event sequences with frequencies of $5x10^{-7}$ per plant year to $1x10^{-4}$ per plant year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification. The objective of BDBEs to assure the capability of the plant is the same as in the NRC definition.
Design Basis Accidents (DBA)	"Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components." [SRP 15.0]	Postulated accidents that are used to set design criteria and limits for the design and sizing of SSCs that are classified as safety-related. DBAs are derived from DBEs and high consequence BDBEs based on the capabilities and reliabilities of safety related SSCs to mitigate and prevent accidents, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SSCs classified as safety related are available to mitigate the consequences to within the 50.34 dose limits.
Licensing Basis Events (LBEs)	Term not used formally in NRC documents	The entire collection of events considered in the design and licensing basis of the plant, which may include one or more reactor modules. LBEs include AOOs, DBEs, BDBEs, and DBAs

Related Questions on LBE Identification

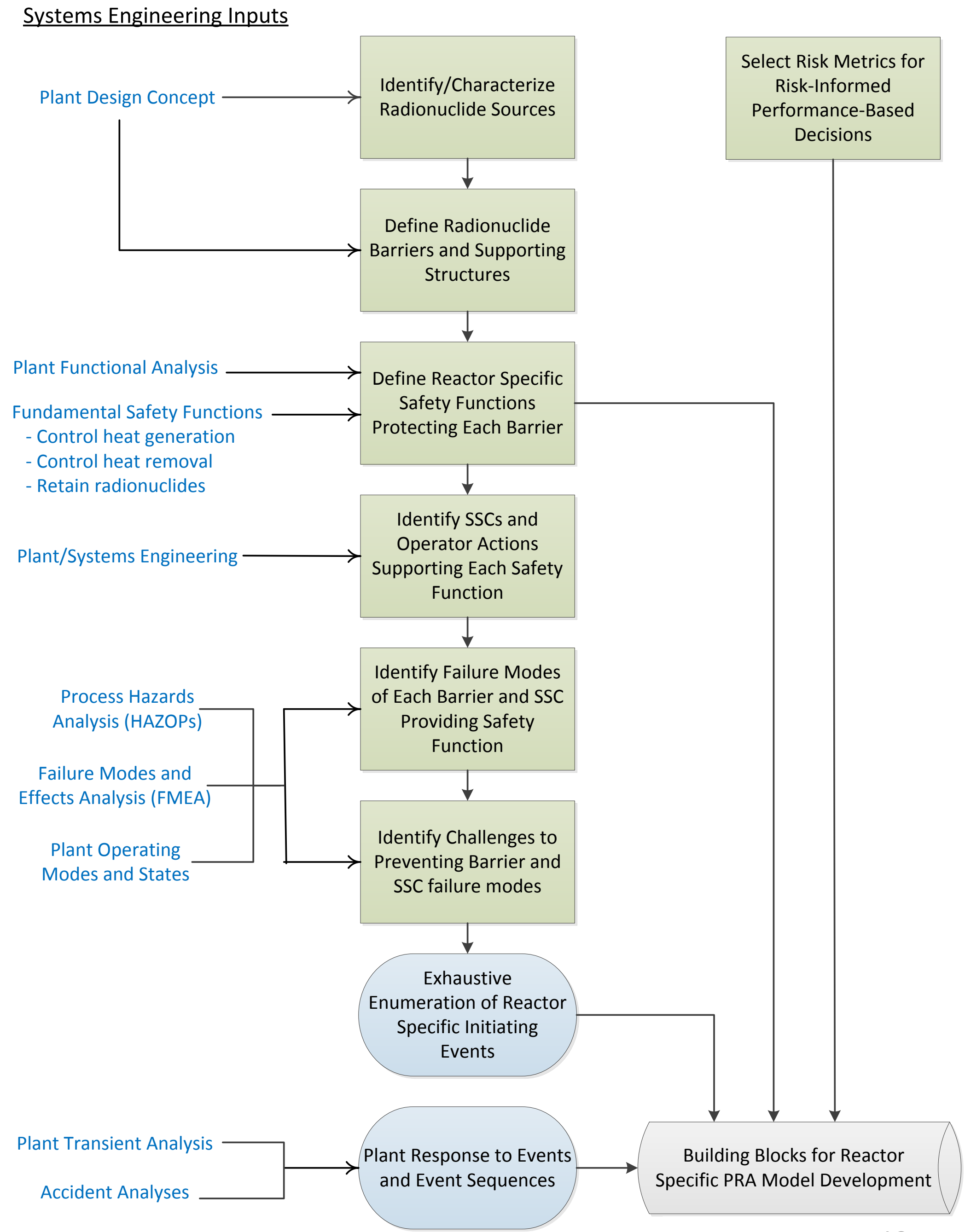
(2) The construct for current LWRs could be described as including both ~~a~~ deterministic analyses (stylized, conservative, barrier-based acceptance criteria) and probabilistic analyses (best estimate, dose-based acceptance criteria), with a balancing of the approaches providing added confidence in designs and operations (assessing from somewhat different perspectives). How might incorporating different approaches to analyses and acceptance criteria be used for advanced reactors to gain similar confidence?	Clarify: "deterministic" in the question and generally. We should recognize that judgement is not deterministic in selecting design events. A qualified calculation or analysis would be a deterministic way to validate the judgement. The existing process has resulted in a prescriptive list of things to assess and rules on how to assess them. Not relying on the SFC is also more robust, ie, systematically looking at all combinations of failures and performance reliability and resilience should add more confidence in the approach, particularly when there are fewer or no operational precedents with a design.
(3) Beyond discussions of "engineering judgement" in defining and assessing LBEs, is it expected that all assessments would include a basic set of events challenging key safety functions of reactivity control, decay heat removal, and limiting the release of radioactive material.	
(7) The NGNP LBE selection approach bins PRA sequences into AOOs, DBEs, and BDBEs based solely on the sequence frequencies. The proposed approach will produce AOOs and DBEs that are different in character than have traditionally been defined. For example:	Clarify -Is this a question of LBEs v DBA?. LBEs in the DBE region may still be events that are dependent on non-SR SSCs. The SR SSC-only events may be in the BDBE region. this is a good thing as it shows DID and robustness, particularly if the DBA is still below the PAG. Variations on that situation need to be talked through.
7b. When defining DBEs, it seems necessary to first identify the SSCs whose performance is to be assessed using the DBE, then map relevant PRA sequences to the identified SSCs. The 10^{-4}/y frequency boundary between AOOs and DBEs seems higher than used for current LWRs.	Clarify. Are we talking about DBAs or LBEs? There are no LBEs per se in existing LWR licenses.

Process For Selecting and Evaluating LBEs

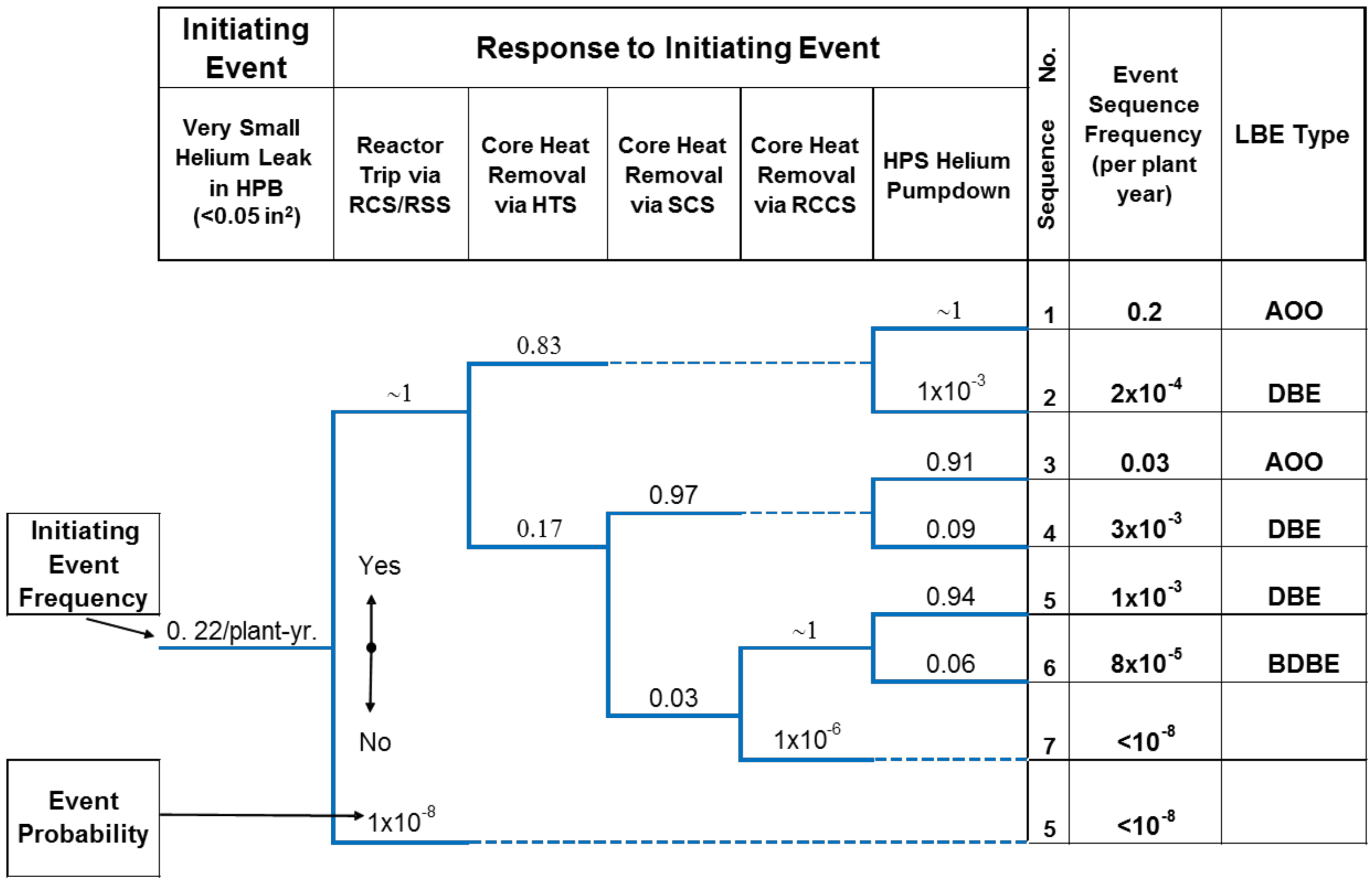


Flow Chart for Initial PRA Model Development

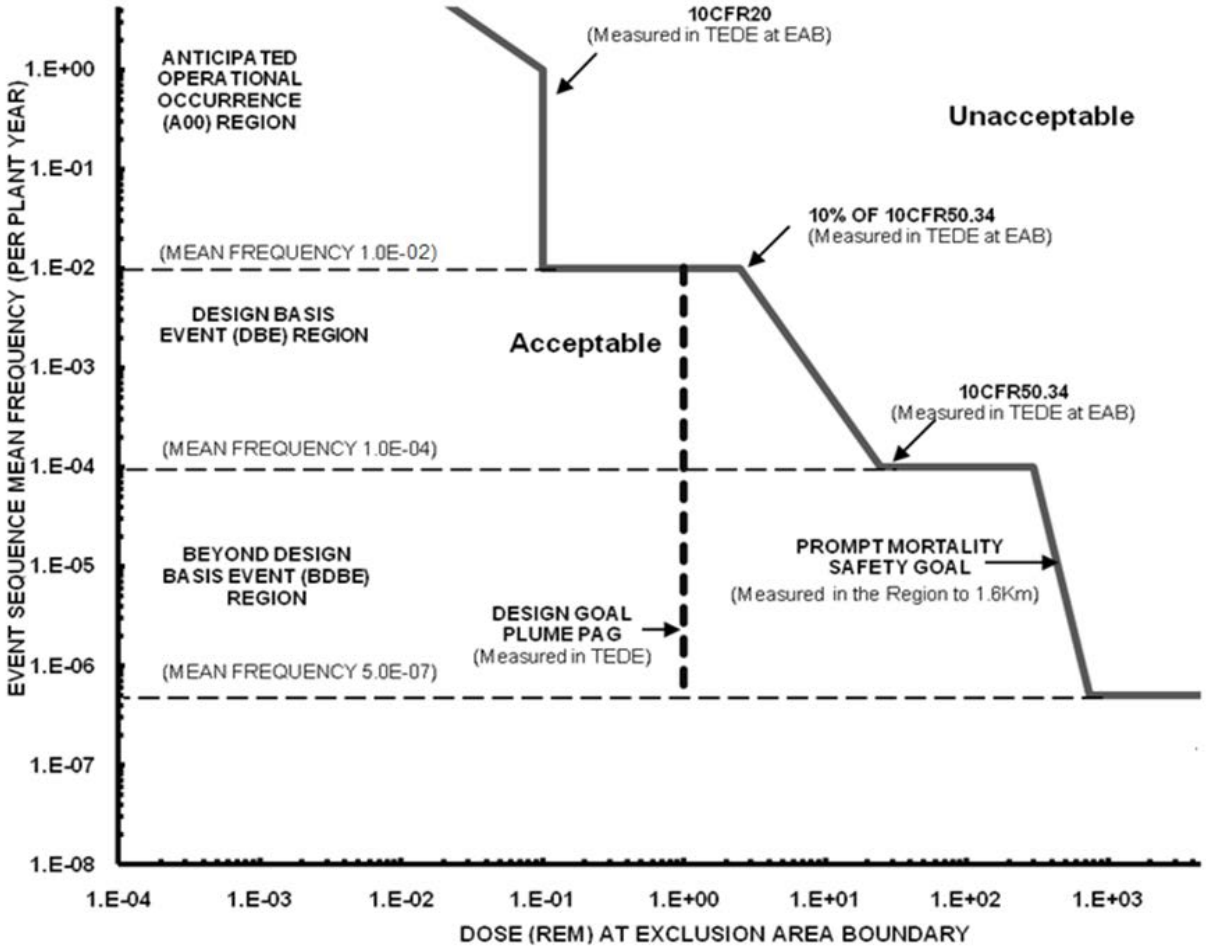
- Expands on Steps 2 and 3
- Risk metrics and criteria for LBE evaluation defined in Step 7
- Focus is on early stages of design and PRA development



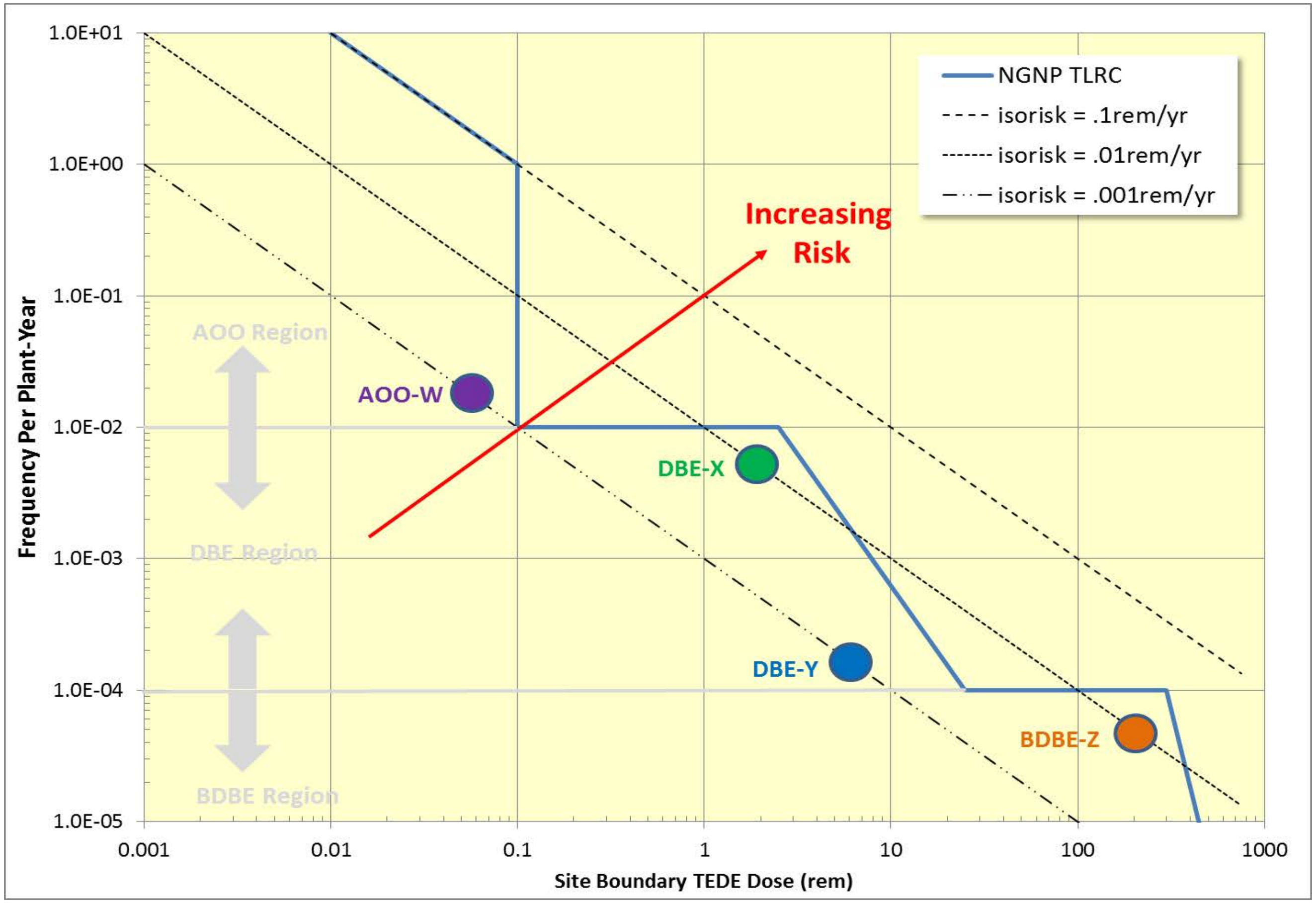
Event Tree for MHTGR Very Small Leaks in Helium Pressure Boundary



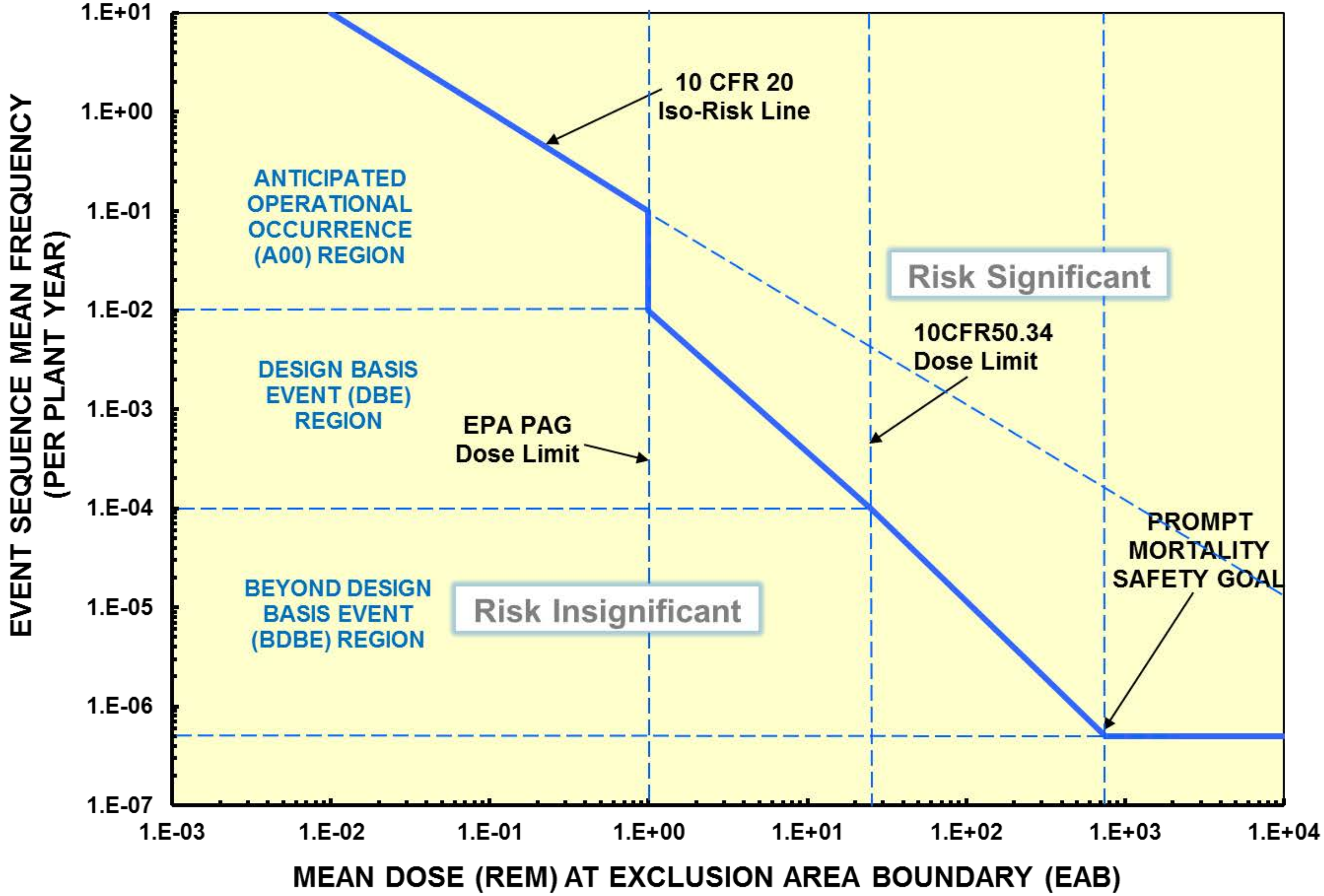
NGNP TLRC Frequency – Consequence Evaluation Criteria



“Staircase Discontinuity Issue”



Example Frequency-Consequence Evaluation Criteria Without Staircase Issues



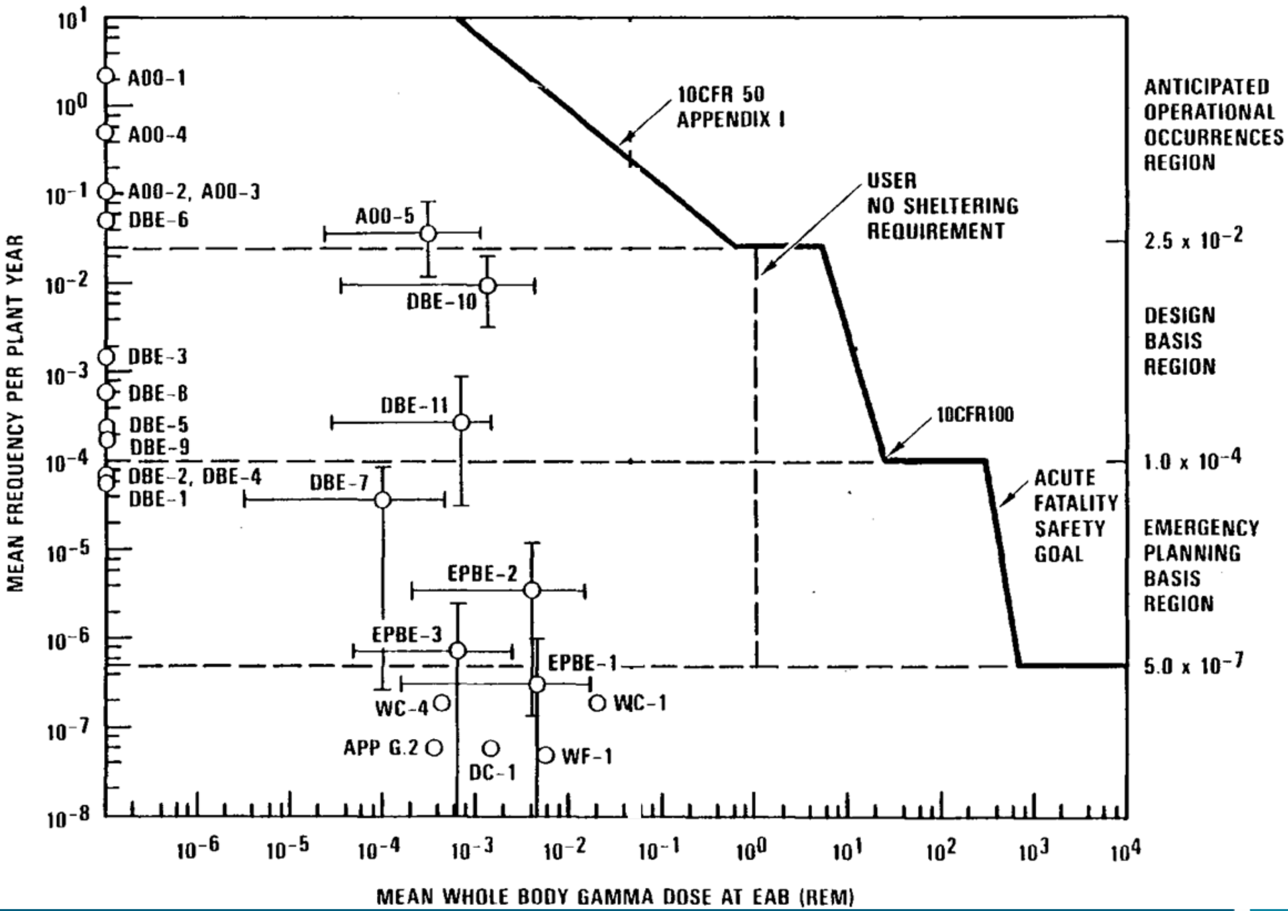
Related Questions on LBE Evaluation

(10) Dose & Siting Questions Intro : The regulations on safety analysis information related to design and siting for construction permits and operating licenses under Part 50 and Design Certifications, COLs, Standard Design Approvals and Manufacturing Licenses under Part 52, (10 CFR 50.34(a)(1) and 10 CFR 52.47(a)(2)(iv), 52.79(a)(1)(vi), 52.137(a)(2)(iv), and 52.157(d), respectively), require design basis accident radiological consequences analysis that includes an evaluation of safety features and the barriers that must be breached before a release of radioactive material to the environment can occur. The regulation further states that this analysis shall assume a large fission product release from the core into the containment, and an evaluation and analysis of postulated fission product release using the demonstrable containment leak rate and any release mitigating systems to evaluate the offsite radiological consequences (dose at EAB and LPZ). DBA radiological consequence analyses for large LWRs have included such following features – a standardized assumed release to the containment based on full core melting without vessel breach, assumed release from the containment is based on leak rate tested through technical specifications surveillance program (La at Pa), credit for only ESF SSCs unless non-safety-related SSCs make the radiological release or consequences greater, other release pathways including estimated potential leakage from liquid containing systems outside of the containment, 95th percentile atmospheric dispersion coefficients for the specific site (i.e., dispersion is worse only 5% of the time resulting in radioactive material concentration at dose receptor locations at the higher end of projected values) or a site parameter used in design certifications, standard design approvals and manufacturing licenses, and the evaluation of radiological consequences of atmospheric release (plume) resulting in the calculation of maximum 2-hr dose at EAB and dose at LPZ for duration of the plume passage. Control room radiological habitability analysis uses the same DBAs as evaluated for offsite doses, and includes the evaluation of the control room habitability SSCs.	Clarify per discussion
10.2. Does the DBA dose siting and safety analysis fit in the overall licensing basis event classification, or is it a separately postulated analysis or set of analyses ?	
10.3. Considering the current regulatory requirements how will the DBA dose siting and safety analysis **assumptions** (accident scenario, transport modeling, fission product removal modeling, etc.) be determined for advanced reactors?	
10.4. a) How do you envision that core damage frequency, release frequency or scenario likelihood would play a role in the selection of the DBA for radiological consequence assessment for siting and safety analysis?	

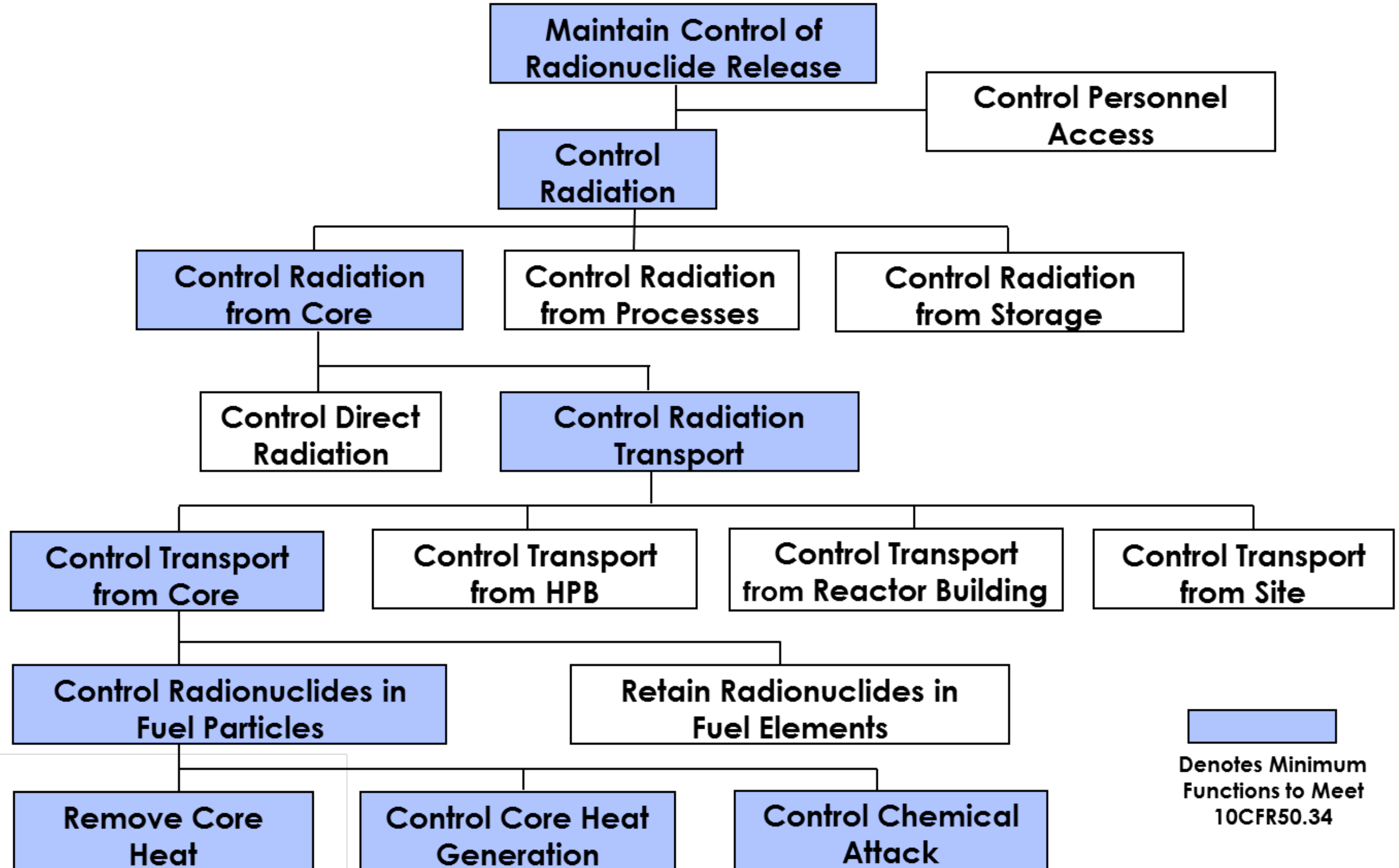
Risk Evaluation of LBEs

- Purpose of TLRC is to evaluate the risk significance of individual LBEs
- Integrated plant risks must also satisfy cumulative risk criteria
  - QHO for early fatality individual risk
  - QHO for latent cancer fatality individual risk
  - Safety Goal for Large release frequency
  - Annual exposure limits per 10 CFR Part 20

Risk Evaluation of MHTGR LBEs vs. TLRC Frequency – Dose Criteria



MHTGR Safety Functions (partial) Including Those Required to Meet 10 CFR 50.34 Limits



Evaluation of MHTGR SSCs for Core Heat Removal Safety Function

Alternate Sets of SSCs	Design Basis Events									SSCs Classified as SR?
	DBE 1	DBE 2	DBE 3	DBE 4	DBE 5	DBE 6/7	DBE 8/9	DBE 10	DBE 11	
Reactor HTS ECA	No	No	No	No	No	No	No	No	No	No
Reactor SCS SCWS	No	Yes	Yes	No	Yes	Yes	Yes	Yes	No	No
Reactor RV RCCS	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**	**Yes**
Reactor RV RB	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No

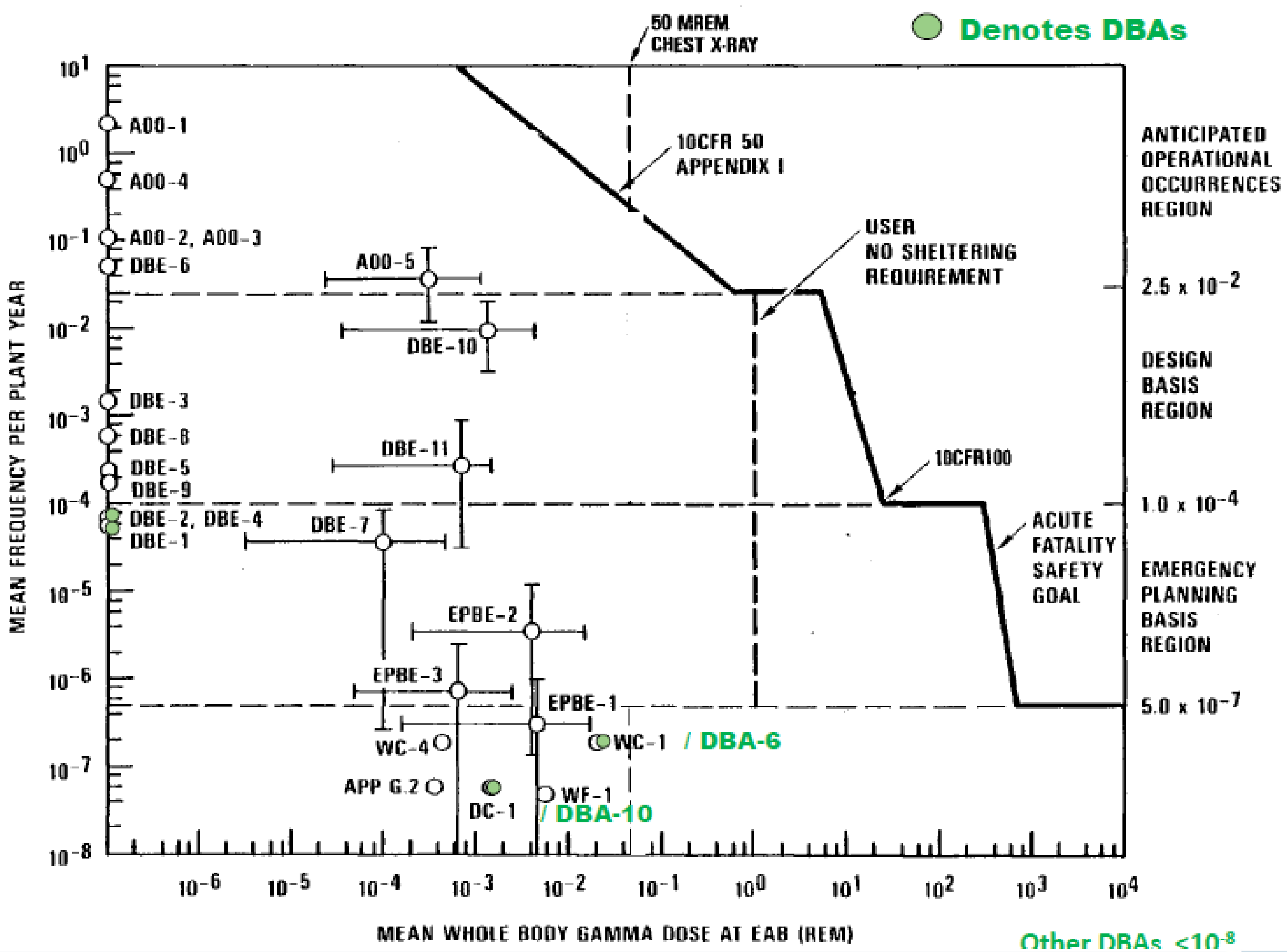
Examples of MHTGR DBE and Associated DBAs

DBE	Design Basis Events	DBA	Design Basis Accidents
DBE-1	Loss of offsite power initiating event and SCS forced cooling, successful reactor trip, passive cooling via RCCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of 5x10^{-5}/plant-year or about 1x10^{-5}/reactor-year)	DBA-1	Loss of Main and SCS forced cooling, successful reactor trip, passive cooling via RCCS, intact HPB and no release involving a single reactor module (corresponds to PRA sequence family with frequency of 5x10^{-5}/plant-year or about 1x10^{-5}/reactor-year)
DBE-2	Main Loop Transient with Control Rod Trip failure, successful reactor trip via RSS, forced cooling via SCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of 7x10^{-5}/plant-year or about 2x10^{-5}/reactor-year)	DBA-2	Loss of Main and SCS forced cooling with Control Rod Trip failure, successful reactor trip via RSS, passive cooling, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of 7x10^{-5}/plant-year or about 2x10^{-5}/reactor-year)
DBE-3	Control Rod Withdrawal, with successful reactor trip, Main Loop forced cooling failure, forced cooling via SCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of 2x10^{-3}/plant-year or about 5x10^{-4}/reactor-year)	DBA-3 DBA-4	Control Rod Withdrawal, with successful reactor trip, failure of forced cooling via Main loops and SCS, passive cooling via RCCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of 7x10^{-5}/plant-year or about 2x10^{-5}/reactor-year)
DBE-4	Control Rod Withdrawal with successful reactor trip, loss of Main and SCS forced cooling via failures, passive cooling via RCCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of 7x10^{-5}/plant-year or about 2x10^{-5}/reactor-year)		
DBE-5	Seismic event with loss of offsite power, successful reactor trip, continued forced cooling via Main ~~Loops~~ or SCS, intact HPB and no release involving all four reactor modules. (corresponds to PRA sequence family with frequency of 2x10^{-4}/plant-year or 2x10^{-4}/reactor-year)	DBA-5	Seismic event with loss of offsite power, successful reactor trip, failure of forced cooling via Main ~~Loops or~~ and SCS, passive cooling via RCCS, intact HPB and no release involving all four reactor modules. (corresponds to PRA sequence family with frequency of 6x10^{-8}/plant-year or about 6x10^{-8}/reactor-year)
DBE-6	Moderate SG leak with successful reactor trip, SG isolation and dump, forced cooling via SCS, intact HPB and no release involving a single reactor module. (corresponds to PRA sequence family with frequency of 5x10^{-2}/plant-year or about 1x10^{-2}/reactor-year)	DBA-6	Moderate SG leak with successful reactor trip and SG isolation, failure of SG dump, failure of forced cooling via SCS, passive cooling via RCCS, circulating activity and delayed fuel release via primary relief valve to reactor building involving a single reactor module. (corresponds to PRA sequence family with frequency of 2x10-7/plant-year or 5x10^{-8}/reactor-year.)
DBE-10	Moderate HPB leak with successful reactor trip, continued forced cooling, release of circulating activity and lift-off of plateout to reactor building involving a single reactor module. (corresponds to PRA sequence family with frequency of 1x10^{-2}/plant-year or about 3x10^{-3}/reactor-year)	DBA-10	Moderate HPB leak with successful reactor trip, failure of forced cooling via Main ~~loops~~ and SCS, passive cooling via RCCS, release of circulating activity, delayed fuel release, and lift-off of plateout to reactor building involving a single reactor module. (corresponds to PRA sequence family with frequency of 6x10^{-8}/plant-year or about 1.5x10^{-8}/reactor-year)

Related Questions on LBE Completeness

7a.1 · The design basis accident for evaluating ECCS performance in LWRs is a large LOCA, followed by an assumed loss of offsite power and the worst-case single failure. This sequence does not result in core damage, and its frequency is well below 10^{-4}/y.	See next item
7a.2 · The design basis accident for an LWR containment assumes (in accordance with 10 CFR 50.34) a fission product release into the containment due to a substantial meltdown of the core. The frequency of this accident is less than 10^{-4}/y.	Clarify: Are there features of the LMP approach that by using a more structured, mechanistic process,the ability to evaluate functional containment performance cannot be satisfied for severe accidents?
(9) a) In the iterative process of using PRA in the design (i.e., PRA insights are fed into the design), how are the uncertainties addressed?	
9 b) How is the design process used to address the known uncertainties?	
10.1. Do advanced reactors propose to comply with the regulations on DBA dose siting and safety analysis, or are exemptions under consideration ?	Clarify: some analysis focused regulations are prescriptive and specific to LWR.s. This may be a nonLWR gap where the solution on how to address the underlying principles applicable to nonLWRs should be discussed
10.5. How will advanced reactors that may not have leak-tight pressure retaining containments propose to comply with the current siting and safety analysis regulations with respect to the assumptions on release to containment and subsequent release from the containment at a rate that is able to be demonstrated over the life of the facility?	

Example MHTGR LBEs, DBAs on F-C Plot



Future Topics for Discussion

(1) a). Describe how performance requirements would be defined for SSCs beyond those required to limit the releases below the F/C curve.	Partially defer: detailed discussion of how SSC classification for all events is done and performance requirements are decomposed as more and more detailed system and component performance-based criteria are developed. will be contained in the future SSC paper.
(5) The scope of the PRA used to support the LBE selection approach includes all hazard groups (internal events and external events such as seismic, flooding, etc.):	
5a. Currently, NRC accepts the use of a PRA-based seismic margins analysis (not a seismic PRA, which is necessary to estimate seismic sequence frequencies and consequences).	Clarify: How do the OBE, SSE and BDB earthquake values get established for or used in the PRA? Are there other terms that are needed to have a rationalized RI exeternal events initial event set? Is the use of Seismic PRA not acceptable for a new design or is SMA necessary?
5 b. How to determine an appropriate reference site for external hazard frequencies ?	
(8) The non-LWR PRA standard is a high level standard and tells the user what is needed, and how to implement.	
a) Since the PRA is going to have a much bigger role, how will the applicant and NRC ensure there is technical adequacy?	Clarify: Is this more about QA or technical adequacy?
b) Does this change the nature of the peer review?	

Future Topics for Discussion

b) If barrier based or other surrogate measures are being used to define performance measures for specific SSCs, would such alternate measures be defined and become part of the licensing basis for the subject SSCs.	Clarity - barriers when mentioned can be considered "functional" barriers. - surrogate measures are not being proposed. Are there some instances where surrogate measures are necessary?
c) Could the logic include meeting the F/C curve if an explicit safety criterion for a barrier has been met (this approach described in a Toshiba 4S report).	Clarify- are 4S barriers functional vs physical. Using barrier integrity as a surrogate may not be appropriate for all LBE conditions. Some events of interest may bypass a barrier, Some designs may not have traditional barriers, eg. Liquid fuel MSRs. Were 4S barriers prescibed or derived from existing design features?
(4) a)How would the proposed approach change the treatment of the PRA and the content of applications? Currently the PRA is described in Chapter 19 in addition to the deterministic evaluations in Chapter 15.	Defer detailed answer. Implications of RIPB insights to many of the FSAR sections needs to be evaluated. Discuss how NRC would like to get into this topic since the SSC Class, DID, PRA 2, and RIDM papers will all touch on this question . The application would benefit from some refocusing pg Ch 15 on "Plant Capability Analysis". Because of the conservative rules applied to DBA vs. BE performance, separating the nominal performance and ASME, ACI etc . design performance points from the analysis of extreme event capability could benefit the subsequent use of RIPB design and adequacy evaluations and operational RIDM processes
b)How would the combination of the two be reflected in the applications and the treatment of the PRA?	Defer detailed answer. Implications of RIPB insights to many of the sections needs to be evaluated. SSC and DID discussions are all over the FSAR

Future Topics for Discussion

(6) a) What are the appropriate quality requirements for thermal-hydraulic codes (analogous to MAAP and MELCOR) and consequence analysis codes (MACCS) used to the support a PRA that forms the basis of the LBE selection?	Clarify: role of RG1.231 for supporting codes. A discussion with NRC re quality of PRA inputs to RIDM.
b)Does RG 1.203, "Transient and Accident Analysis Methods," apply ?	
c) How are safety margins and defense-in-depth addressed?	
10.4. b) How does the selection of the DBA consider defense-in-depth or other factors not related to risk assessment?	Defer: The DID paper will elaborate on the DID evaluation methodology and DID adequacy criteria
10.6. For advanced reactor designs that can acceptably demonstrate that radiological release through core damage events is not physically possible, how would the design meet current regulatory requirements ?	Clarify: this is more a licensing question, not a safety question. It would meet the TLRC and the PDC developed for the design. This would comply with the TI performance based regulations. The prescriptive LWR-centric regulations would (should) not be applicable to that advanced design.
10.7. Is there a desire to maintain references to the existing DBA siting criteria and include the EPA PAG limit as a goal, or might the EPA PAG limit at design-specific distances be used as a more established design limit?	Clarify: The choice between a goal and a limit depends on the implications of each and the benefits to the developer/owner and regulator. Are there ways to maximize the benefits without minimizing design flexibility that could be adverse to non-regulatory developer and owner objectives?

QUESTIONS?