Advisory Committee on Reactor Safeguards (ACRS)
Future Plant Designs Subcommittee

10 CFR Part 53
“Licensing and Regulation of
Advanced Nuclear Reactors”

December 16-17, 2021

December 16th Agenda

9:30am – 9:40am	Opening Remarks & Staff Introductions
9:40am – 10:00am	Update on Part 53 Rulemaking Schedule
10:00am – 1:00pm	Graded PRA and Possible Licensing Pathways
1:00pm – 2:00pm	Lunch Break
2:00pm – 5:00pm	Staffing, Operator Certification, Simulators [Subpart F - Requirements for Operations]
5:00pm – 5:30pm	Discussion

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NRC Staff Plan to Develop Part 53

Subpart B | **Subpart C** | **Subpart D** | **Subpart E** | **Subpart F** | **Subpart G**

Requirements Definition
- Safety Objectives
- Safety Criteria
- Safety Functions

← Project Life Cycle →

Design and Analysis	Siting	**Construction**	Operation	Decommission
System & Component Design	*External Hazards*	*Construction/ Manufacturing*	*Configuration Control*	*Funding*
Analysis Requirements	*Site Characteristics*	*Ensuring Capabilities/ Reliabilities*	*Surveillance Maintenance*	*Termination*
Safety Categorization & Special Treatment	*Environmental Considerations*	*Change Control*	*Staffing & Human Factors*	
		Environmental Considerations	*Programs Security, EP*	
			Facility Safety Program	

Other
- **Subpart A** General Provisions
- **Subpart J** Admin & Reporting
- **Other 10 CFR Parts**

Plant/Site (Design, Construction, Configuration Control) →

Analyses (Prevention, Mitigation, Compare to Criteria) →

Plant Documents (Systems, Procedures, etc.) →

LB Documents (Applications, SAR, TS, etc.) **Subparts H & I** →

Clarify Controls and Distinctions Between

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Current Status

Subpart	Subpart Description	Status
A	General Requirements	Released **1st** iteration, including initial definitions (April 2021)
B	Safety Criteria	Released **3rd** iteration (August 2021)
C	Design and Analysis	Released **3rd** iteration (August 2021)
D	Siting	Released **1st** iteration (April 2021)
E	Construction	Released **1st** iteration (April 2021)
	Manufacturing	Released **1st** iteration (April 2021)
F	SSCs	Released **1st** iteration (April 2021)
	Personnel	Released **1st** iteration (October 2021)
	Programs	Released **1st** iteration (April 2021)
G	Decommissioning	Under development (Planned release **December 2021**)
H	Licensing (LWA, ESP, SDA, DC)	Released **1st** iteration (August 2021)
	Licensing (ML, CP/OL, COL)	Released **1st** iteration (October 2021)
I	Maintaining Licensing Basis	Released **1st** iteration (August 2021)
J	Reporting & Financial	Released **1st** iteration (August 2021)
Part 5X	Deterministic Alternative	Released **1st** iteration (October 2021)
Part 73	Physical Security	Released **2nd** iteration (November 2021)
	Cyber Security	Released **2nd** iteration (November 2021)
	Access Authorization	Released **2nd** iteration (November 2021)
Part 26	Fitness-for-duty	Under development (Planned release **December 2021**)
Other	Technology-Inclusive, Risk-Informed Maximum Accident Approach	Under development
	Conforming Changes	Under development
	Statements of Consideration	Under development
	Regulatory Analysis	Under development

Revised Part 53 Rulemaking Schedule

Key Rulemaking Milestones	Activity Date(s)
Public Outreach & Generation of Proposed Rule Package	July 2020 to August 2022
• *NGO Public Meeting*	*February 2022*
Submit Draft Proposed Rule to Commission	February 2023
Commission Review	March 2023 to May 2023
OMB and OFR Processing	May 2023
Publish Proposed Rule	**June 2023**
Public Outreach & Generation of Final Rule Package	June 2023 to May 2024
Submit Draft Final Rule to Commission	December 2024
Commission Review	January 2025 to April 2025
OMB and OFR Processing	April 2025 to June 2025
Publish Final Rule	**July 2025**

Other Key Milestones & Activities

- Continuation of topical public meetings, as needed.
- Continuation of frequent ACRS meetings
  - Focus of the meetings will be topics ACRS members want to discuss in more detail
- Continuing to publicly release preliminary proposed rule text
  - In January 2022, the staff will release a section of the preliminary FRN with all of the Part 53 subparts in one document

Coordination with Other Rulemakings

- Emergency Planning for Small Modular Reactors (SMRs) and Other New Technologies (ONTs)
- Limited Scope Physical Security
- Decommissioning
- Part 50-52 Lessons Learned
  - Severe Accidents (add to Part 50)
  - Probabilistic Risk Assessments (PRAs) (add to Part 50)
  - Three Mile Island (TMI) Requirements (add to Part 50)
  - Fire Protection Requirements (align Parts 50 and 52)
  - Operating Licensing (revise simulation, walkthrough requirements)
  - Licensing Processes (eliminate need to renew design certifications (DCs) and standard design approvals (SDAs))
  - Environmental (allow reference to DC environmental assessment)

Graded PRA and Possible Licensing Pathways

Marty Stutzke
Division of Advanced Reactors and Non-Power and Utilization Facilities
Office of Nuclear Reactor Regulation

December 16, 2021

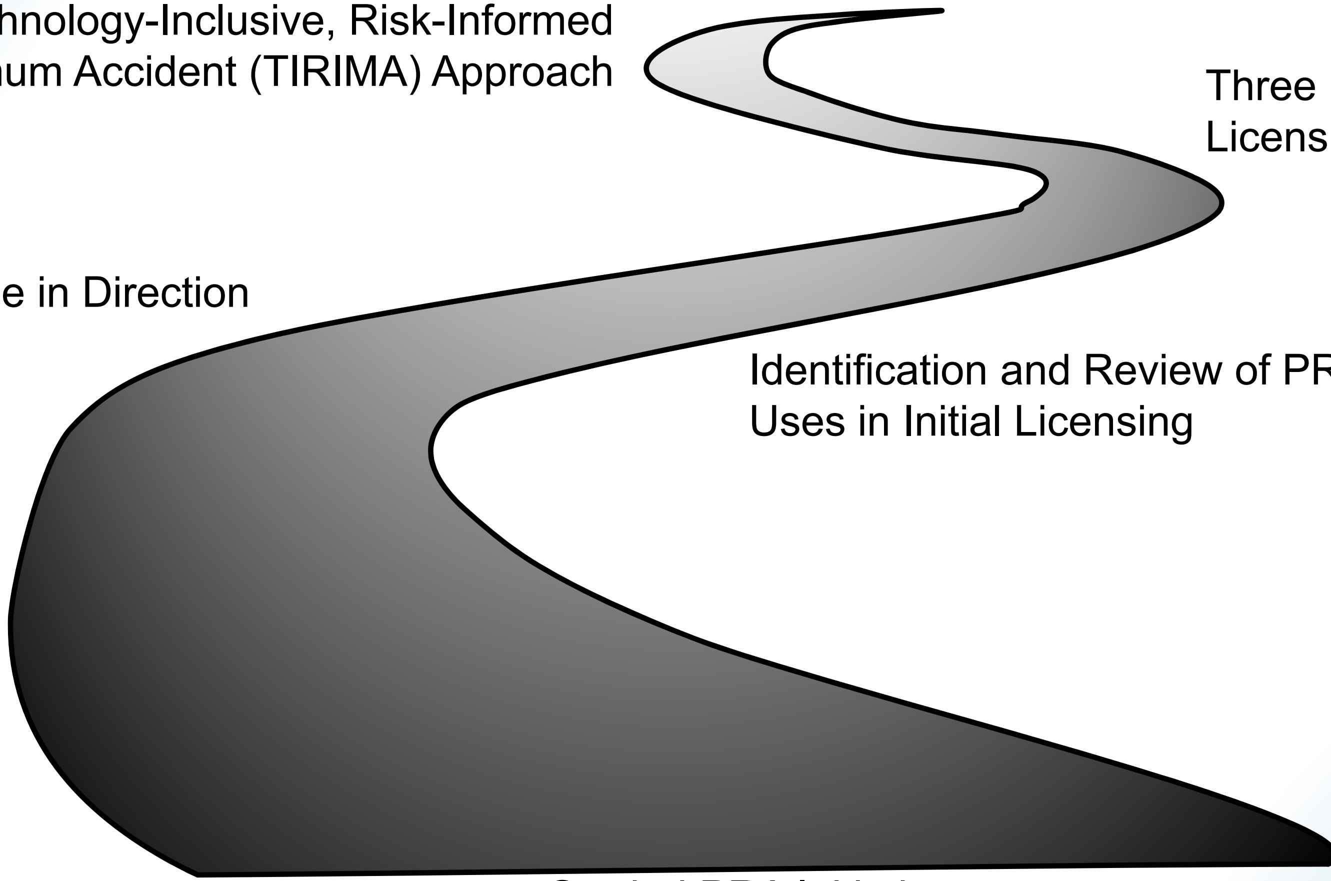
Overview

Technology-Inclusive, Risk-Informed Maximum Accident (TIRIMA) Approach

Three Possible Licensing Pathways

A Change in Direction

Identification and Review of PRA Uses in Initial Licensing



Graded PRA Initiative

Graded PRA Working Group

- In the spring of 2021, a working group was formed to develop viable options to “grade” (“right-size” or “customize”) the PRA developed to support initial license applications (construction permits (CPs), operating licenses (OLs), DCs, SDAs, manufacturing licenses (MLs), combined licenses (COLs)).
- The staff originally envisioned a three-phase process:

Phase 1 — Develop a graded PRA approach

Phase 2 — Craft guidance

Phase 3 — Explore alternatives to PRA

Spring 2021 Working Definitions

Graded PRA approach

- A process that uses bounding, conservative, and/or qualitative assessments to establish a PRA's scope, level of detail, degree of plant representation, and/or level of peer review commensurate with the licensing stage (which dictates the level of detail and finality of the information used to develop the PRA) and how the PRA will be used in risk-informed decision-making.

Graded PRA

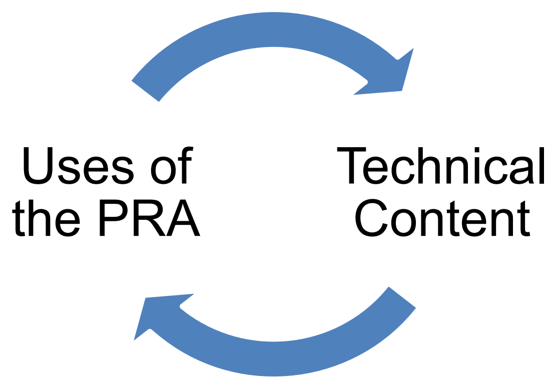
- A PRA of appropriate degree of scope, level of detail, plant representation, and technical adequacy to support a specific advanced reactor licensing application.
- Note: "Graded" should not imply that a design is not yet complete –acceptance of a graded PRA could only be considered if a design is well understood and conservatively modeled.

Dose/consequence-based criterion

- A potential entry condition to enable a graded PRA that uses bounding, conservative, and/or qualitative assessments of the doses or consequences arising from potential unplanned release scenarios, without consideration of the release scenario likelihood. This approach is being considered as a specific criterion for developing a graded PRA to adequately demonstrate that an applicant meets the intent of the Commission's Severe Accident Policy in an efficient and effective manner.

A Change in Direction

- Based on feedback during the Advanced Reactor Stakeholders public meeting held 5/27/2021, the staff learned that industry concerns were largely directed at grading how PRA was used in the licensing process, rather than grading the technical content of the PRA itself.
- There was general recognition from industry that the non-light water reactor (non-LWR) PRA standard already offers opportunities to grade the content of the PRA.



Grading PRA Technical Content: Use of Qualifiers

- The non-LWR PRA standard (ASME/ANS RA-S-1.4-2021) uses qualifiers to indicate when certain supporting requirements (SRs) apply.
- Examples:
  - o "For operating plants, …"
  - o "For PRAs performed during the pre-operational stage, …"
- SRs without qualifiers apply to all life cycle stages.

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Grading PRA Technical Content: The Risk Application Process

Section 3 of the non-LWR PRA standard (ASME/ANS RA-S-1.4-2021) provides a process to grade the PRA:

- Describes required activities to establish the capability of a PRA to support a particular risk-informed application.
- Designed to be used during all life cycle stages.

Stage A
- Characterize plant life cycle stage and PRA application
- Define site characteristics
- Select PRA scope and level of detail
- Select risk significance criteria
- Determine Capability Category needed to for each part of the PRA needed to support the application

Stage B
- Determine if PRA scope, level of detail, and risk metrics are sufficient to support the application
- Upgrade PRA as needed

Stage C
- Determine if PRA standard requirements are sufficient for application.
- Determine if insufficient requirements are relevant to the application.

Stage D
- Determine if PRA has sufficient capability to support application
- If not, determine if deviations are significant to the applicant
- Upgrade the PRA as needed

Stage E
- Use PRA to support the application
- As needed, use supplementary analyses and requirements to support the application

> Staff will provide PRA-related guidance for initial licensing in TICAP and/or ARCAP guidance.

Focus on Understanding How and Why PRA is Used

- Since the May 2021 Advanced Reactor Stakeholders public meeting , the staff has further explored the scope of the PRA and how it is used in initial licensing.
- Significant effort has been invested in thoroughly understanding:
  - The uses and role of the PRA in the licensing process,
  - Whether those uses and role could be adequately addressed with other tools/techniques/bounding assessments, and
  - How that information fits into the overall approach to licensing under Part 50, Part 52, and preliminary Part 53.

Required Uses	Expected Uses
• Regulations • Rulemakings	• Regulatory guides (RGs) • Commission policy statements • Commission staff requirement memoranda • Standard review plans • IAEA SSR-2/1

Why PRA? Post-TMI Recommendations

- ACRS letter* (May 16, 1979)
  - The ACRS believes that it is time to place the discussion of risk, nuclear and non-nuclear, <u>on as quantitative basis as possible</u>.
- Kemeny Report* (October 30, 1979) Recommendation #4:
  - The [Presidential] Commission recommends that continuing in-depth studies should be initiated on the <u>probabilities and consequences</u> (on-site and off-site) of nuclear power plant accidents, including the consequences of meltdown.
- Rogovin Report* (NUREG/CR-1250, January 1980), Recommendation #8:
  - The best way to improve the existing design review process is by relying in a major way upon <u>quantitative risk analyses</u>, and by emphasizing those accident sequences that contribute significantly to risk.

*Available from the Idaho National Laboratory Knowledge Management Library for the Three Mile Island Unit 2 Accident of 1979 at https://tmi2kml.inl.gov/HTML/Page1.html

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Advanced Reactors and PRA

Regulations and Rulemakings

- 2019: Part 53 rulemaking
- 2009: Parts 50/52 lessons learned rulemaking; propose to add PRA requirements for CP and OL applications
- 8/28/2007: Part 52 revised with requirement to submit a description of the PRA and its results
- 4/18/1989: Part 52 issued with requirements to meet § 50.34(f) and submit PRA
- 1/15/1982: TMI requirements added in § 50.34(f); perform PRA to seek improvements in the reliability of core and containment heat removal systems

Policy Statements

- 10/14/2008: Rev. 2 to ARPS; cites SAPS, SGPS, and PRAPS
- 8/16/1995: PRA policy statement (PRAPS)
- 7/12/1994: Rev. 1 to ARPS; added reference to metrification policy statement
- 8/21/1986: Safety goal policy statement (SGPS)
- 7/8/1986: Advanced reactor policy statement (ARPS); cites SAPS and the forthcoming SGPS
- 8/8/1985: Severe accident policy statement (SAPS); use PRA to search for severe accident vulnerabilities

Advanced Reactor Policy Statement

(73 FR 60612; October 14, 2008)

- Comment labeled “Toshiba-3” (pp. 60613-60614):
  - Comment:
    - Policy statement makes no mention of the use of PRA.
    - Helpful to provide advanced reactor designers with interim guidance regarding NRC efforts for a risk informed, technology neutral licensing framework.
  - Response:
    - The NRC has established specific requirements related to the use of PRA in licensing new nuclear power plants, such as 10 CFR 52.47 and 10 CFR 50.71(h).
    - The Commission has also issued policy statements on:
      - Use of PRA in regulatory activities (PRAPS)
      - Severe accidents regarding future designs and existing plants (SAPS)
- Page 60616: The Commission also expects that advanced reactor designs will comply with the Commission’s SGPS.

PRA Policy Statement

(60 FR 42622; August 16, 1995)

- “The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.”
- “PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices.”
- “PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.”
- “The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.”

Severe Accident Policy Statement

(50 FR 32138; August 8, 1985)

- Describes the policy the Commission intends to use to resolve safety issues related to reactor accidents more severe than design basis accidents (DBAs).
  - Main focus is on the criteria and procedures the Commission intends to use to certify new designs for nuclear power plants.
  - “Severe nuclear accidents are those in which substantial damage is done to the reactor core whether or not there are serious offsite consequences.”
- Policy for new plant applications:
  - “Comply with current regulations, including the TMI requirements in 50.34(f)”
  - “Demonstrate technical resolution of all applicable Unresolved Safety Issues and the medium- and high priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems.”
  - “Complete a PRA and consider the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety.”
  - “Complete a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.”

Safety Goal Policy Statement
(51 FR 28044; August 4, 1986 as corrected and republished at 51 FR 30028; August 21, 1986)

- Qualitative goals
  - "Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health."
  - "Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks."
- Quantitative objectives
  - "The risk to an average individual in the vicinity of a nuclear power plant [one mile] of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed."
  - "The risk to the population in the area near a nuclear power plant [10 miles] of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."
- Proposed general performance guideline (large release frequency – LRF):
  - "Consistent with the traditional defense-in depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

Large LWR Risk Surrogates

- NUREG-0880, Rev. 1, May 1983 (ML071770230), provides numerical interpretation of the quantitative health objectives (QHOs):
  - Individual early fatality risk (IEFR) ≤ 5E-7/ry
  - Individual latent cancer fatality risk (ILCFR) ≤ 2E-6/ry
- NUREG-1860, Vol. 2, December 2007 (ML080440215), derives surrogate risk metrics for large light water reactors (LWRs):
  - IEFR: Large early release frequency (LERF) ≤ 1E-5/ry
  - ILCFR: Core-damage frequency (CDF) ≤ 1E-4/ry

The non-LWR PRA standard (ASME/ANS RA-S-1.4-2021) does not use risk surrogates such as CDF, LRF, or LERF!!!

Large Release Frequency

- SRM-SECY-89-144, June 5, 1990, ML051660712:
  - "Within a particular design class (e.g., LWRs, LMRs, HTGRs) the same subsidiary objectives [risk surrogates] should apply to both current as well as future designs…However, the Large Release Guideline relates to all current as well as future designs."
  - "The Commission believes that "adequate protection" is a case by case finding based on evaluating a plant and site combination and considering the body of our regulations."
- According to Forrest Remick (former Director of the Office of Policy Evaluation, ACRS member, and Commissioner), LRF was proposed to break a deadlock between the ACRS and the staff over the use of CDF and conditional containment failure probability (CCFP) as safety goals (see ML051660709):
  - Staff wanted to only include CDF; ACRS wanted to also include CCFP.
  - Dr. Remick received a call from the Executive Assistant to the then Commission Chairman [Palladino]. "[H]e asked me what I thought of getting out of the deadlock by eliminating both the CDF and the idea of a CCFP by substituting a Large Release Guideline of 10^{-6} per reactor year as a surrogate. In response, I laid out the following steps: if one assumes a CDF of 10^{-4}, a conditional probability of "core on the floor" of 10^{-1} (a probability still not accurately known), and a CCFP of 10^{-1} (which was bandied about at the time), then a Large Release Guideline of 10-6 per reactor year appeared to be a reasonable surrogate."
- SRM-SECY-12-0081, October 22, 2012, ML12296A158: "The Commission has approved the staff's recommendation (Option 2C) to transition from large release frequency to large early release frequency (LERF) at or before initial fuel load and discontinue regulatory use of large release frequency (LRF) and conditional containment failure probability thereafter."
- SECY-13-0029, March 22, 2013, ML13022A207:
  - "The Commission acknowledged that analyses indicated that the cancer fatality QHO was not the more controlling objective and that, if the prompt fatality QHO is met, the cancer fatality risk generally would be much lower than the cancer fatality QHO."
  - "Recognizing that the prompt fatality QHO is more controlling, the staff worked to develop a large release definition that focused on LRF being a surrogate for the prompt fatality QHO."
  - "In 1993, the staff concluded that defining large release beyond a simple qualitative statement related to its 10^{-6} per reactor year release frequency (such as is currently contained in the safety goal policy statement) was neither practical nor required for regulatory or design purposes."

Large Early Release Frequency

- The concept of LERF was originally developed by the Electric Power Research Institute, "PSA Applications Guide," TR-105396, August 1, 1995.
  - Use qualitative characteristics to identify large early release sequences
  - Avoids the need to perform source term and radiological consequence calculations
- Adopted by the NRC when RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," was developed in the late-1990s.
- Current definition provided in RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," Rev. 3, December 2020:
  - Large early release frequency (LERF) is defined as the sum of the frequencies of those accidents leading to rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there is the potential for early health effects. (Such accidents generally include unscrubbed releases associated with early containment failure shortly after vessel breach, containment bypass events, and loss of containment isolation.)

Risk-Informed Regulation Definitions (1 of 2)

- Risk triplet (Kaplan and Garrick*, SRM-SECY-98-144**):
  - What can go wrong?
  - How likely is it?
  - What are the consequences?

$$R = \{\langle s_i, p_i(\varphi_i), \zeta_i(x_i)\rangle\}$$

- Risk assessment (SRM-SECY-98-144): A systematic method for addressing the risk triplet as it relates to the performance of a particular system (which may include a human component) to understand likely outcomes, sensitivities, areas of importance, system interactions and areas of uncertainty.
- Risk insights (SRM-SECY-98-144): The results and findings that come from risk assessments.
- Risk-informed approach (SRM-SECY-98-144): A philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety.

*Kaplan, S. and Garrick, B. J., "On the Quantitative Definition of Risk," *Risk Analysis*, Vol. 1, Issue 1, March 1981.
**NRC, "Staff Requirements – SECY-98-144 – White Paper on Risk-Informed and Performance-Based Regulation," February 24, 1998, ML003752593.

Risk-Informed Regulation Definitions (2 of 2)

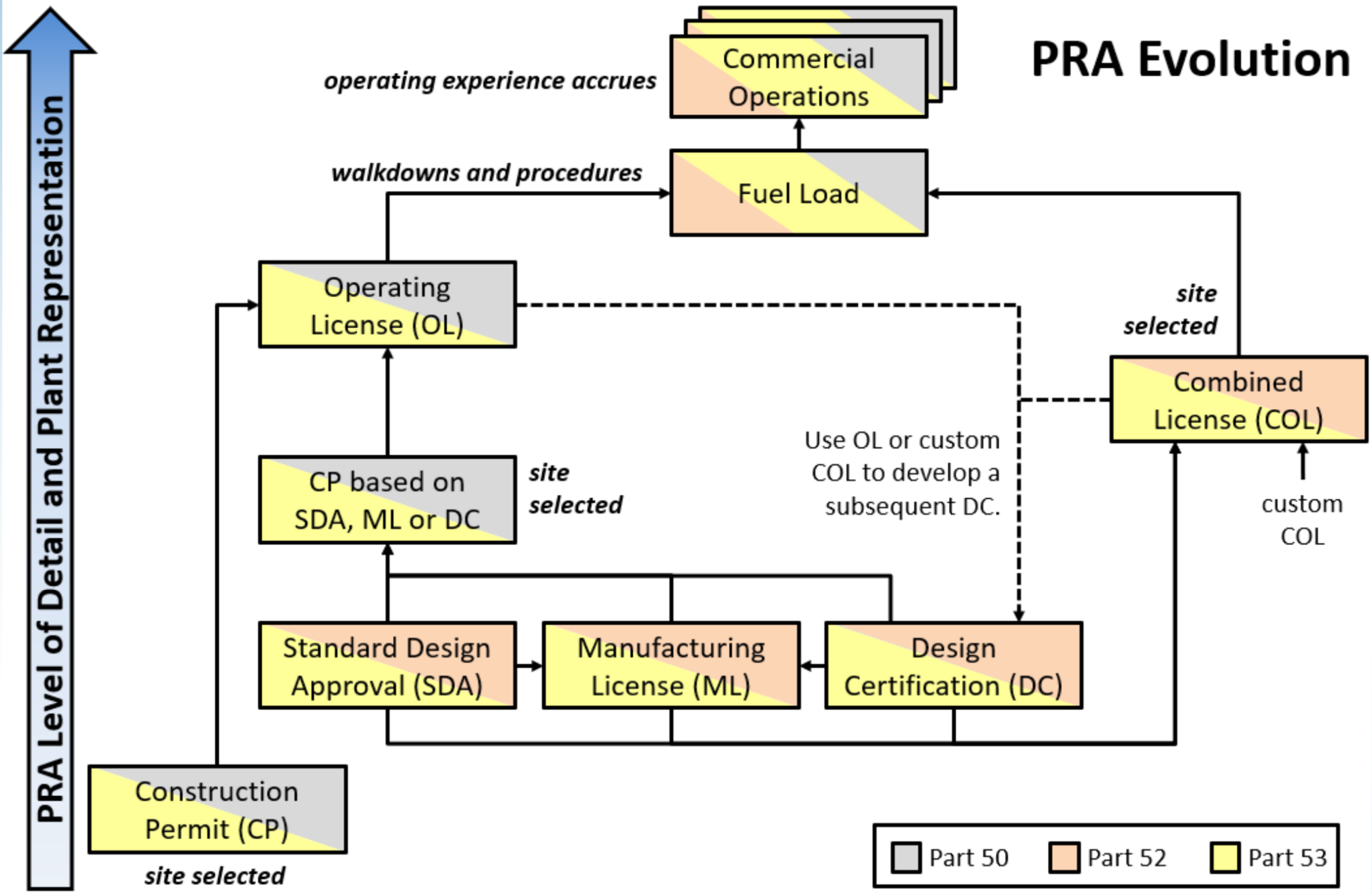
- PRA
  - **NRC online glossary**[*]: A systematic method for assessing three questions that the NRC uses to define "risk." These questions consider (1) what can go wrong, (2) how likely it is, and (3) what its consequences might be. These questions allow the NRC to understand likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty, which the staff can use to identify risk-significant scenarios. The NRC uses PRA to determine a numeric estimate of risk to provide insights into the strengths and weaknesses of the design and operation of a nuclear power plant.
  - **RG 1.200**[**]: An approach is considered to be a PRA when it (1) provides a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequences (e.g., core damage or a large early release) and their frequencies and (2) comprises specific technical elements in performing the quantification.
  - **Draft RG 1.247**[†]: A risk assessment approach is considered to be a PRA when it (1) provides a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequences (e.g., releases of radioactive material, radiological consequences) and their frequencies and (2) comprises specific PRA elements for quantifying risk.

[*] https://www.nrc.gov/reading-rm/basic-ref/glossary/probabilistic-risk-assessment-pra.html
[**] NRC, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Rev. 3, December 2020, ML20238B871.
[†] NRC, "Acceptability of Probabilistic risk Assessment Results for Advanced Non-Light Water Reactor Risk-Informed Activities," draft trial use RG 1.247, September 3, 2021, ML21246A216.

Role of the PRA in Initial Licensing

- Traditional role
  - Consistent with previous DC and COL applications
  - Includes, but not limited to:
    - Searching for severe accident vulnerabilities (SAPS)
    - TMI requirement § 50.34(f)(1)(i), which under Part 52 requires LWR applicants to "Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."
    - Demonstrating that the QHOs are met (SGPS)
    - Using PRA in the design process (PRAPS)
  - Previously referred to as "PRA in a supporting role"
- Enhanced role
  - Any use of PRA beyond its traditional role
  - Includes, but not limited to:
    - Certain proposed required uses of PRA in preliminary 10 CFR Part 53 rule text (e.g., identifying licensing basis events (LBEs); classifying systems, structures, and components (SSCs); evaluating defense-in-depth (DID))
    - Voluntary risk-informed applications (e.g., risk-managed technical specifications (TS), risk-informed fire protection)
  - Previously referred to as "PRA in a leading role"

PRA Evolution



PRA Requirements for Part 50 CP and OL Applicants

- PRA not currently required for CP or OL applicants.
- SRM-SECY-15-0002, ML15266A023: “The Commission has approved the staff’s recommendation to confirm that the Commission’s guidance given in the "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" and other Commission direction identified by staff apply to new 10 CFR Part 50 power reactor applications in a manner consistent with 10 CFR Part 52 design and license applications.”
- The Part 50/52 lessons learned rulemaking (NRC-2009-0196; RIN 3150-AI66) proposes to add PRA-related requirements for CP applicants, OL applicants, and OL holders like the PRA-related requirements for Part 52 applicants and COL holders.

PRA Requirements for Part 52 Applicants

- § 52.47(a)(27) requires DC applicants to describe the design-specific PRA and its results.
- § 52.79(a)(46) requires COL applicants to describe the plant-specific PRA and its results.
  - § 52.79(c)(1) requires COL applicants that reference an SDA to use and update the PRA information for the SDA to account for site-specific design information and any design changes or departures.
  - § 52.79(d)(1) requires COL applicants that reference a DC to use and update the PRA information for the standard DC to account for site-specific design information and any design changes or departures.
  - § 52.79(e)(1) requires that COL applicants that reference an ML to use and update the PRA information for the ML to account for site-specific design information and any design changes or departures.
- § 52.137(a)(25) requires SDA applicants to describe the design-specific PRA and its results.
- § 52.157(a)(31) requires ML applicants to describe the design-specific PRA and its results.

PRA Requirements for Part 52 COL Holders

- § 50.71(h)(1) requires that:
  - Each COL holder shall develop a Level 1 and a Level 2 PRA, and
  - The PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist 1 year before the scheduled date for initial loading of fuel.
- § 50.71(h)(2) requires that:
  - Each COL holder shall maintain and upgrade the PRA
  - The upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect 1 year before each required upgrade, and
  - The PRA must be upgraded every 4 years until the permanent cessation of operations under 10 CFR 52.110(a).
- § 50.71(h)(3) requires that each COL holder shall, no later than the date on which the licensee submits an application for a renewed license, upgrade the PRA required by 10 CFR 50.71(h)(1) to cover all modes and all initiating events.

Subpart C-Design and Analysis Requirements
§ 53.450 Analysis Requirements
(3rd iteration preliminary rule text)

(a) *Requirement to have a probabilistic risk assessment.* A probabilistic risk assessment (PRA) of each commercial nuclear plant must be performed to identify potential failures, susceptibility to internal and external hazards, and other contributing factors to event sequences that might challenge the safety functions identified in 53.230 and to support demonstrating that each commercial nuclear plant meets the safety criteria of 53.220.

Traditional use of the PRA

Severe accident policy statement

Safety goal policy statement

Subpart C-Design and Analysis Requirements
§ 53.450 Analysis Requirements
(3rd iteration preliminary rule text)

(b) *Specific uses of analyses.* The PRA, other generally accepted risk-informed approach for systematically evaluating engineered systems, or combination thereof must be used:

Enhanced use of the PRA

(1) In determining the licensing basis events, as described in 53.240, which must be considered in the design to determine compliance with the safety criteria in Subpart B of this part.

(2) For classifying SSCs and human actions according to their safety significance in accordance with 53.460 and for identifying the environmental conditions under which the SSCs and operating staff must perform their safety functions.

(3) In evaluating the adequacy of defense-in-depth measures required in accordance with 53.250.

PRA scope

- **All sources**
- **All hazards**
- **All modes**

(4) To identify and assess all plant operating states where there is the potential for the uncontrolled release of radioactive material to the environment.

(5) To identify and assess events that challenge plant control and safety systems whose failure could lead to the uncontrolled release of radioactive material to the environment. These include internal events, such as human errors and equipment failures, and external events, such as earthquakes, identified in accordance with Subpart D of this part.

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Uses of the PRA

Requirements/Uses	Part 50	Part 52	Part 53 Preliminary Rule Text
Submit description of PRA and its results	Currently none. Proposed in Part 50/52 lessons learned rulemaking (NRC-2019-0196; RIN 3150-AI66)	All applicants	All applicants
Develop, maintain, and upgrade PRA		COL holders	COL and OL holders
Required uses of PRA		Meet the TMI requirements in § 50.34(f)(1)(i) – Seek improvements in core and containment heat removal systems reliability	Use PRA to: • Search for severe accident vulnerabilities • Demonstrate that safety goals are met • Use PRA to evaluate changes to the facility described in FSAR (§ 53.1322) • Use PRA **or generally accepted risk-informed approaches for systematically evaluating engineered systems** to: ○ Identify LBEs ○ Evaluate DID ○ Classify SSCs ○ Support the FSP
Commission expectations (e.g., policy statements and SRMs)	• Search for severe accident vulnerabilities • Demonstrate that safety goals are met • Use PRA in design		
Voluntary uses of PRA	Voluntary risk-informed applications to establish or change the licensing basis		
Leveraged uses by the staff	• Focus the staff review • Inform the development of ITAACS, COL action items, D-RAP, etc. • Support oversight and inspections		

☐ Traditional role of PRA
☐ Enhanced role of PRA

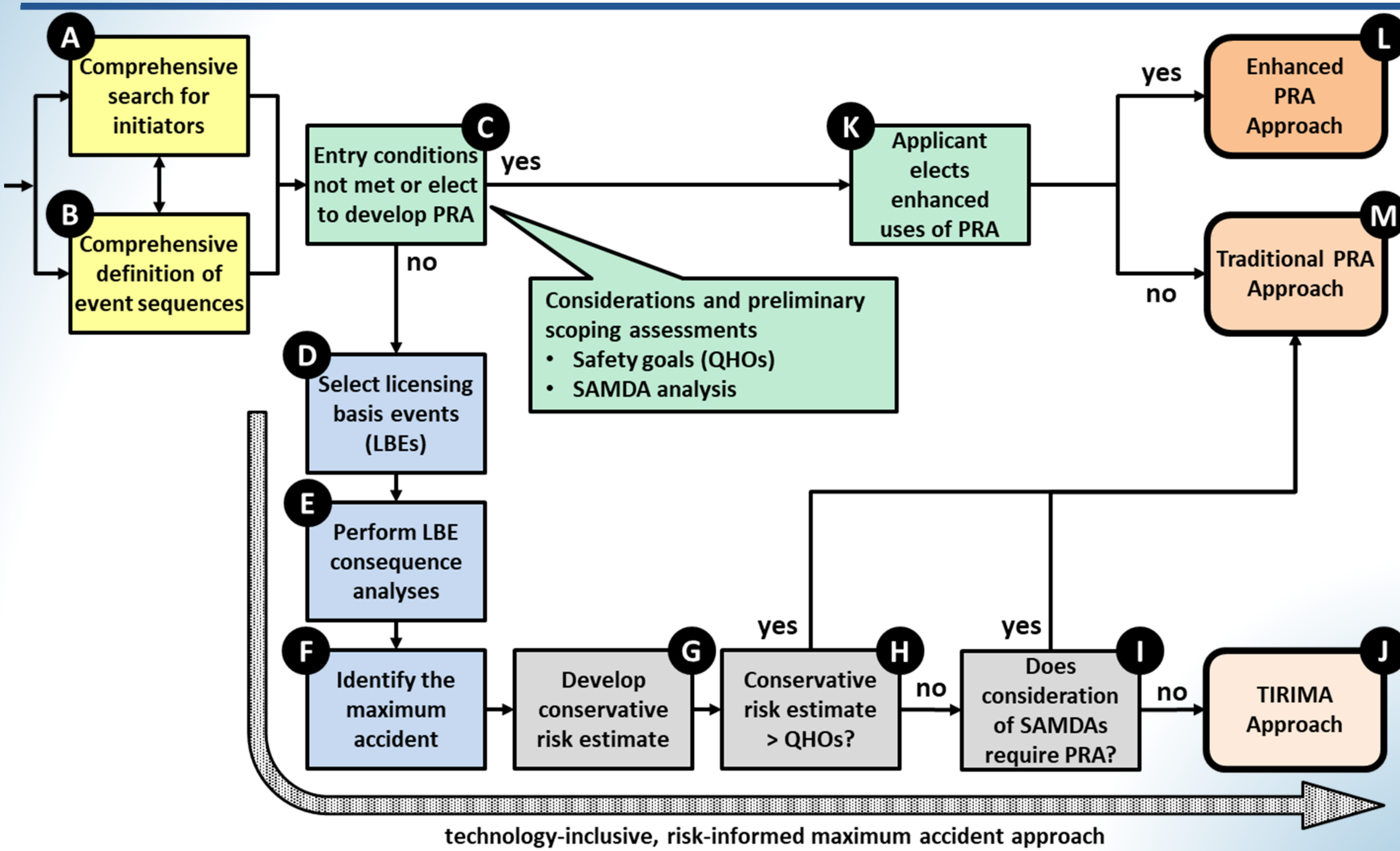
Observations

- A risk-informed approach may be based on risk insights developed from:
  - A PRA (i.e., quantitative), or
  - A qualitative risk assessment
- PRA not used to support non-power production and utilization facility licensing:
  - Not addressed in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors"
  - Discussion with NRR/DANU staff
- Integrated safety analysis is required by 10 CFR Part 70 for certain licensees:
  - Risk-informed process, but does not require development of a PRA
- The current preliminary rule text for Part 53:
  - Builds on the traditional role of the PRA
  - Adds requirements that use PRA in an enhanced role

Three potential licensing pathways

- Enhanced PRA approach
- Traditional PRA approach
- TIRIMA approach

Three Licensing Pathways



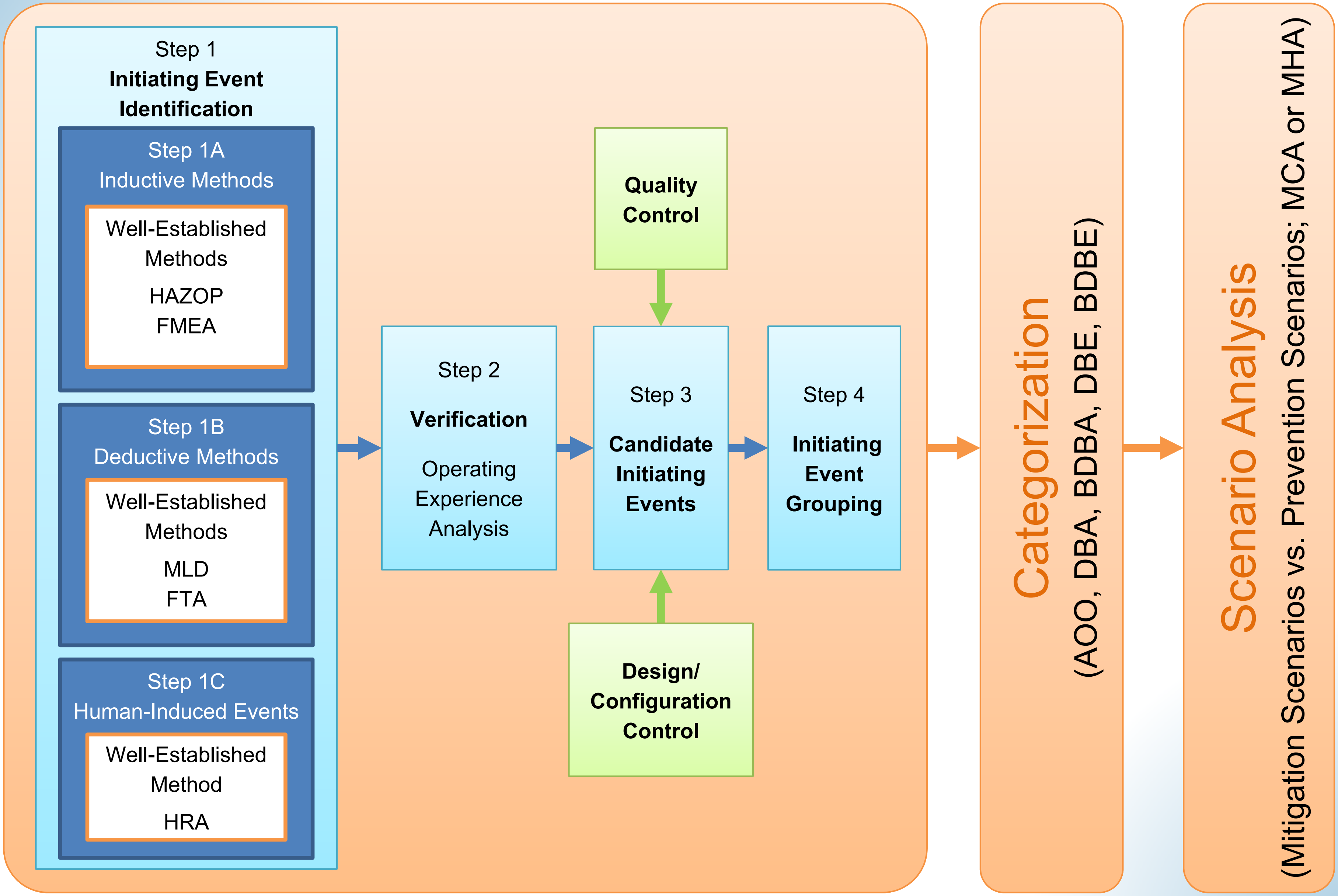
Development of “How-To” Guidance for the TIRIMA Approach (1 of 2)

- <u>The staff intends to develop guidance for</u>:
  - Box A: Comprehensive search for initiating events
  - Box B: Comprehensive definition of event sequences
  - Box C: Decision guidance and entry conditions
  - Box D: Licensing basis event selection
  - Box E: Licensing basis event consequence analysis
  - Box F: Maximum accident identification
  - Box G: Conservative risk estimation
- <u>Leverage existing guidance and studies such as, but not limited to</u>:
  - NUREG-1513, “Integrated Safety Analysis Guidance Document”
  - NUREG-1520, “Standard Review Plan for Fuel Cycle Facilities License Applications”
  - NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors”
  - Occupational Safety and Health Administration regulations (29 CFR 1910.119), standards, handbooks, and guidance
  - EPRI TR 3002011801, “Program on Technology Innovation: Early Integration of Safety Assessment into Advanced Reactor Design - Preliminary Body of Knowledge and Methodology”

Development of "How-To" Guidance for the TIRIMA Approach (2 of 2)

- Initial thoughts:
  - Start with a blank sheet of paper
  - Use a combination of inductive and deductive methods
  - How much searching is enough? How do you know when you are finished?
  - Focus on how plant design actually works vs. how plant design is supposed to work
  - Consolidate/group similar items
  - Be (very) careful when screening
- Process:
  - Multi-disciplinary team effort
  - Independent review
  - Documentation
    - Tell the story
    - Capture assumptions and decisions

Initiating Event and Scenario Search



Initiating Events Analysis
Preliminary Table of Contents

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Path Forward

- Revise preliminary rule text developed for the "deterministic option" to allow use of the TIRIMA approach
- Develop guidance to implement the TIRIMA approach:
  - Leverage existing guidance to the extent possible
  - Outline of guidance to conduct the systematic and comprehensive search for initiators (Box A) has been developed
  - Developing outline of the entire guidance document
- Challenging technical issues:
  - Technology-inclusive definition of "severe accident"
  - Providing guidance to help applicants decide up-front if TIRIMA is a viable licensing pathway (TIRIMA entry conditions – Box C)
  - Using TIRIMA to support SAMDA analysis

TIRIMA Approach

Discussion

MEETING BREAK

Meeting to resume in 1 hour

Subpart F – Requirements for Operation

Subpart F – Requirements for Operation

- Operational objectives
- Transition from construction/manufacturing to operation
- Maintaining capabilities and availability of SSCs
- Maintenance, repair, and inspection programs
- Design control
- Staffing, training, personnel qualifications, human factors requirements
- Programs

Subpart F - Programs

- Radiation protection
- Emergency preparedness (EP)
- Security programs
- Quality assurance
- Integrity assessment programs
- Fire protection
- Inservice inspection/inservice testing (ISI/IST)
- Criticality safety program
- Facility safety program (FSP)

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Overview of Topics for Discussion

- Shift Technical Advisor (STA) Position
- Load Following
- Certified Operators versus Licensed Operators
- Simulator Scope
- Training Program Review Guidance

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Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Shift Technical Advisor Position – Background

§ 53.753(f) requires a staffing plan describing numbers, positions, and qualifications of licensed operators and senior licensed operators, or certified operators, across all modes

- Must describe personnel providing support in plant operations, equipment surveillance and maintenance, radiological protection, chemistry control, fire brigades, engineering, security, and emergency response.
- Must also describe how the proposed licensed operator staffing, for plants requiring licensed operators, will be sufficient to assure that plant safety functions can be maintained; this must be supported by human factors engineering (HFE) analyses and assessments.
    - Guidance for evaluating these staffing plans is being developed by the staff in the form of interim staff guidance (ISG) to be used in conjunction with NUREG-1791

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Shift Technical Advisor Position

- Staff have considered various approaches to the STA role under Part 53 and several alternatives will be discussed here
- Staff are receptive to feedback on different approaches as further iterations of preliminary rule language are developed
- In developing preliminary Part 53 requirements, the staff considered that the 1985 Policy Statement on engineering expertise on shift (50 FR 43621) stated that the STA was an interim measure until goals that included upgrading human-system interfaces (HSIs) and operator training were achieved
    - o Current perspective is that the upgrades to HSIs and operator training envisioned within this Policy Statement will be the norm under Part 53 and driven by regulatory requirements
- Staff recognize that this represents a policy issue and intend to use the Part 53 rulemaking process for Commission engagement

Shift Technical Advisor Position (cont'd)

Generic Part 53 elimination of STA could be justified on following bases:

- Licensed operator training requirements on knowledge and abilities to maintain plant safety functions; review criteria would confirm testing of reactor theory, thermodynamics, systems, and emergency operations
- State-of-the-art HFE required in all settings where operators are fulfilling plant safety functions; design requirements for HSIs requiring operators be provided with information on safety parameters, safety system status, important component status, and core damage states
- HFE-based analyses and assessments required to demonstrate how licensed operator staffing levels will maintain safety functions and support full range of tasks needed for safety (irrespective of an STA)
- Part 53 codifies DID principles under § 53.250 and requires (in part) DID use compensates for uncertainties in state of knowledge and modeling capabilities, and for personnel reliability and performance

Shift Technical Advisor Position (cont’d)

Staff have also considered alternative of codification of an STA staffing requirement, with provision for justifying case-by-case position omission:

- In such an approach, the following definition would be provided in § 53.750(b):

  Shift technical advisor means an individual possessing at least a baccalaureate in physical science, engineering, or engineering technology (or, alternately, a Professional Engineering license) and whose function is to provide independent on-shift engineering expertise, accident assessment, and technical advice to licensed operators at nuclear power plants.

- The staffing plan requirements of § 53.753(f) would be modified to include the following additional requirement for plants requiring licensed operator staffing:

  A description of how the shift technical advisor position, as defined by § 53.750(b), will be implemented during all plant conditions other than cold shutdown or refueling while shutdown or, alternatively, shall provide a justification for omission of the Shift Technical Advisor position that is supported by relevant human factors engineering-based analyses and assessments.

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Shift Technical Advisor Position (cont'd)

- Another alternative considered by staff would require that proposed staffing plans for facilities with licensed operators account for how on-shift engineering expertise will be provided
  - Such an approach would likely be accomplished by modifying the requirements of § 53.753(f) such that facilities requiring licensed operators would need to describe (in part) within staffing plans:
    - how the numbers and positions of licensed operators provide assurance that plant safety functions can be maintained, and
    - how on-shift engineering expertise will be provided for the on-shift crew
- Overall, the staff perspective currently remains that any STA position requirement would only apply to plants that require licensed operators

Load Following – Background

§ 53.755, "Conditions for Operations Staffing for Operating or Combined Licenses under this Part"

- § 53.755(c) restricts control manipulations to licensed or certified operators.
- § 53.755(e) requires that operations (other than control manipulations) affecting reactor power level only occur while plant conditions are being monitored by a licensed or certified operator.
  - However, load-following is permitted if one of the following can immediately refuse demands from the grid operator when they could challenge safe operation or if precluded by equipment conditions:
    - the actuation of an automatic protection system,
    - an automated control system; or
    - a licensed or certified operator.

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Load Following (cont'd)

- Current intent is to supplement rule with guidance (e.g., ARCAP ISG)
- While preliminary, the following illustrates general staff perspectives:
  - Load following should be restricted to power levels where automation supports needed plant operations in order to avoid transients (e.g., if one feedwater train must be secured as part of reducing power <50% and automation cannot accomplish this, then 50% becomes the bottom of the load following envelope)
  - If the actuation of an automatic protection system will be relied upon to truncate/terminate load following, then any such protective actions should not be the same as those credited for core protection and should use more restrictive setpoints than those credited for safety purposes (avoids challenging limits)
  - Crediting operators requires an immediate capacity to take control
  - No restriction envisioned on facility types if an adequate design is demonstrated; usage of certified operators could be acceptable

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Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Certified Operators – Background

- § 53.755(a) requires facility licensees to have licensed operators unless they can meet criteria of § 53.755(b) to use certified operators
- § 53.755(b) contains the requirements that must be met in order to justify not using any licensed operators as a part of facility staffing
  - Two current proposals for criteria
  - The first proposal would require the following:
    - No human actions for event mitigation required to meet safety criteria, achieve safety functions, or provide DID
    - PRA demonstrating the evaluation criteria for each event sequence can be met without human action for mitigation;
    - LBE response not needing human action for SSCs to perform
  - The second proposal would require the design-basis accident safety criteria to be met without mitigation by human actions, active engineered features, or passive design features (except for only those passive features that can survive LBEs and not be defeated by credible human errors)

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Certified Operators – Background (cont'd)

- § 53.755(i) contains specific requirements for plants using certified operators
  - o Certified operator are responsible for specified administrative functions.
  - o Certified operator staffing must always ensure continuity of responsibility for facility operations during the operating phase.
  - o Continuous monitoring of fueled units with the following capabilities:
    - receiving plant operating data and parameters
    - ability to immediately initiate a reactor shutdown
    - ability to promptly dispatch ops/maintenance personnel
    - the ability to implement any emergency plan responsibilities
    - conducting reactivity manipulations that require human action

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Certified Operators – Background (cont'd)

- § 53.774, "Issuance of Certificates" (for Certified Operators)
  - Requires that facility licensees ensure that individuals to be certified:
    - complete either a high school diploma or GED
    - complete the approved initial training program
    - pass an initial operator certification examination
    - demonstrate competence in conducting control manipulations
    - meet medical condition requirements

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

<u>Certified Operators – Background (cont'd)</u>

- § 53.775, "Conditions of Certificates" (for Certified Operators)
  - Requires facility licensees to ensure that certified operators:
    - only perform duties at facilities for which they are certified
    - complete a continuing training program
    - pass periodic continuing training examinations
    - Complete biennial medical examinations
    - maintain proficiency in accordance with the facility program

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Purpose of Certified Operator Alternative

- If a facility lacks an operator role in safety (e.g., an autonomous reactor design), then a key driver warranting federal licensing of individuals is removed (i.e., operator performance would not have a credible influence on public health and safety outcomes within that context)
    - Regardless of whether the operators were licensed, the facility itself would still be licensed by the NRC
- Important administrative job tasks that would remain still need to be accomplished by adequately qualified personnel.
    - Precedent shows that similar administrative tasks have been fulfilled by non-licensed personnel, such as Certified Fuel Handlers

Purpose of Certified Operator Alternative (cont'd)

- Staff are endeavoring to create a durable rule; part of this is accounting for the possibility of future advancements in safety
- While the framework for certified operators parallels that for licensed operators, there is less regulatory interface and more flexibilities; this brings with it the potential for cost savings on the part of the industry
  - Current industry burden estimates for staff fee-billable hours suggest such differences may reduce costs

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Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Certified Operators versus Licensed Operators

Comparison of licensed and certified operator program components:

Program Component	Licensed Operator	Certified Operator
NRC approval of training & exam programs required?	Yes	Yes
NRC approval of exams prior to administration?	Yes	No
NRC approval of operator applications & medical?	Yes	No
NRC approval of simulators for use in training & exams?	Yes	No
Required submittal of renewals & terminations?	Yes	No
NRC approval of examination waivers?	Yes	No
Flexibility for requalification training & exam periodicity?	No	Yes

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Certified Operators versus Licensed Operators (cont'd)

- Certified operators would be trained to conduct reactivity manipulations as part of their initial training program
- Due to the non-licensed nature of certified operators, the facility licensee retains ultimate accountability for operations
  - In contrast, licensed operators are individually accountable
  - Certified operators have a reduced operator safety role due to facility safety characteristics
- § 53.753(e) requires an operating experience program; staff envision guidance for licensed and certified operator training programs will include the incorporation of operating experience
- There are no known industry initiatives for maintaining Part 53 programs consistent within facility classes; staff are working to ensure compliance with Atomic Energy Act (AEA) requirements
- If a facility licensed with certified operators was determined to later need licensed operators (e.g., the § 53.755(b) criteria were no longer met), a safety issue may exist; NRC would have authority to modify the facility's license as necessary

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Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

<u>Simulator Scope – Background</u>

- § 53.765(e) establishes simulation facility requirements for plants with licensed operators and § 53.773(e) establishes less stringent simulation facility requirements for plants with certified operators
  - o Full-scope simulators are not mandated; partial scope simulators may be acceptable, provided that the scope is adequate to meet intended usage; alternatives to simulators are possible as well
  - o Simulation facilities for plants with licensed operators must be approved by the Commission if the facility licensee will rely upon them for training, experience requirements, or for initial or requalification examinations
    - Equivalent approval not required for certified operator facilities
  - o Must demonstrate that adequate simulator scope is provided to support HFE analyses/assessments in order to use a simulation facility for conducting these analyses/assessments

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Simulator Scope

- In developing preliminary rule language, staff reviewed Section 306 of the Nuclear Waste Policy Act (NWPA) and 52 FR 9453 which discussed implementation of the Act's simulator-related provisions:
  - Flexibilities were historically provided to allow for potential use of the plant itself, and/or a plant-referenced simulator, and/or some other type of simulation device (such as a part-task or basic-principles simulator) for the conduct of the simulator portion of the operating test
  - The NRC's stated intent was not to permit the initiation of transients on the plant itself if used as a simulation facility; rather, the use of the plant was envisioned as an option that might be used in conjunction with another simulation device or devices, in lieu of a plant-referenced simulator
- Current perspective is that NWPA does not mandate NRC to require that plants have *simulators,* but instead requires regulations address the use of *simulations* in training; flexibility exists to allow the use of the actual plant to "simulate" tasks for training and operating test purposes without having a separate simulation facility (e.g., simulator)

Simulator Scope (cont'd)

- Philosophical basis behind preliminary rule language is:
  - Plant-referenced, full-scope simulators remain the preferred approach and would represent the best route for meeting Part 53 requirements
    - Staff expect majority of Part 53 applicants will have them due to regulatory certainty and technology lowering the associated costs
  - Existing regulations do not strictly mandate plant-referenced, full-scope simulators either, but still adopted by all current power reactors
  - Part 53 rule language leaves alternatives to simulator usage (full-scope or otherwise), but the burden will be on the applicant to demonstrate how the following are supported:
    - Licensed or certified operator training and exams; simulators used require sufficient scope and fidelity for operators to acquire and demonstrate knowledge and abilities needed for job duties.
    - Experience requirements (i.e., reactivity manipulations)
    - HFE analyses/assessments and HSI design testbed needs

Training Program Review Guidance – Background

- § 53.765(a) requires initial licensed operator training programs to:
  - Be based upon a systems approach to training (SAT)
  - Ensure that license applicants at the facility will possess the knowledge, skills, and abilities necessary to:
    - protect the public health, and
    - maintain design-specific plant safety functions
  - Be approved by the Commission prior to use
- § 53.765(c) requires facilities to establish requalification training programs for licensed operators. These programs must:
  - Be based on SAT
  - Ensure that reactor operators and senior reactor operators maintain knowledge, skills, and abilities necessary to protect the public health and maintain those plant safety functions specific to the facility design.
  - Be conducted for continuous period not to exceed 24 months.
  - Be approved by the Commission.

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Training Program Review Guidance (cont'd)

- § 53.773(a) requires initial operator certification training programs to:
  - Be based upon SAT
  - Ensure that certified operator trainees will possess the knowledge, skills, and abilities necessary to protect the public health
  - Be approved by the Commission prior to use
- § 53.773(c), requires continuing training programs for certified operators to:
  - Be based upon SAT
  - Ensure that certified operators maintain the knowledge, skills, and abilities necessary to protect the public health
  - Be approved by the Commission prior to use
- §§ 53.780-781 addresses training requirements for other plant personnel
  - § 53.781, "Training and Qualification Requirements"
    - Requires use of SAT
    - Requires the training and qualification of supervisors, technicians, and other appropriate operating personnel to be provided for

Subpart F – Staffing, Training, Personnel Qualifications, and Human Factors Requirements

Training Program Review Guidance (cont'd)

- Applicants might forgo training program accreditation, requiring:
  - o Staff to determine acceptability of proposed training programs
  - o Staff assessment of ongoing conformance of facility licensee training programs with applicable regulatory requirements;
  - o Staff inspections of other training programs required by § 53.781
- Staff will require guidance to support determinations regarding whether SAT is adequately applied; existing guidance (e.g., NUREG-1220) is dated
  - o Working group established to develop this guidance for both initial and continuing training with objective to have an ISG developed no later than 2024 to support Part 53 rulemaking
    - Effort underway to accelerate development to support near-term applicants applying under Parts 50/52 if needed

Subpart F – Requirements for Operation

Discussion

Final Discussion and Questions

December 17th Agenda

9:30am – 9:35am	Opening Remarks
9:35am – 11:30am	Update on TICAP/ARCAP Guidance Document Developments
11:30am – 11:45am	Break
11:45am – 1:00pm	U.S. Nuclear Energy Institute / U.S. NIC Presentation: Letter of November 5, 2021 and Attachments
1:00pm – 2:00pm	Lunch Break
2:00pm – 3:50pm	U.S. Nuclear Energy Institute / U.S. NIC Presentation (continued)
3:50pm – 4:30pm	Discussion

Advanced Reactors

Overview of ARCAP Roadmap ISG and TICAP DG White Papers

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Purpose

- Provide the ACRS Future Plant Designs Subcommittee an update on the Advanced Reactor Content of Application Project (ARCAP) and Technology Inclusive Content of Application Project (TICAP) guidance document developments since the last Subcommittee briefing in July of 2021
  - Highlight some of the key draft white paper guidance with particular attention to portions of the guidance that maybe of interest to the ACRS
- Key documents associated with this presentation are available on NRC's Advanced Reactor ARCAP/TICAP public webpage (see: https://www.nrc.gov/reactors/new-reactors/advanced/details.html#advRxContentAppProj)

ARCAP/TICAP Background

- Previous ACRS Subcommittee briefings provided:
  March 17, 2021, and July 21, 2021
- ARCAP/TICAP
  - Purpose is to develop technology-inclusive, risk-informed and performance-based application guidance
  - Developed to support 10 CFR Part 50, Part 52, and Part 53 applications
    - Near-term need to develop guidance to support expected advanced reactor Part 50/52 applications using the licensing modernization project (LMP) process
    - LMP process endorsed in RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and performance-based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors”
    - ARCAP/TICAP guidance will be revised as Part 53 proposed rule language is adjusted

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ARCAP/TICAP Background

- TICAP guidance
  - LMP process is used to define guidance for the content of major portions of the Safety Analysis Report (SAR)
    - LMP process uses risk-informed, performance-based approach to select LBEs, develop SSC categorization, identify special treatments for SSCs, and ensure DID adequacy
  - Industry developing key portions of TICAP guidance for NRC endorsement
  - Industry TICAP guidance will be supplemented by NRC staff-developed guidance as necessary
- ARCAP guidance
  - Broader in nature than TICAP and intended to cover guidance for SMR and non-LWR applications for a COL, CP, OL, DC, SDA, or ML.
  - Encompasses TICAP guidance and provides supplemental and additional guidance for SAR and application requirements beyond the SAR.

ARCAP and TICAP - Nexus

<u>Outline SAR – Based on TICAP Guidance</u>

1. General Plant Information, Site Description, and Overview of the Safety Case
2. Methodologies and Analyses
3. LBE Analysis
4. Integrated Evaluations
5. Safety Functions, Design Criteria, and SSC Safety Classification
6. Safety-Related (SR) SSC Criteria and Capabilities
7. Non-safety-related with special treatment (NSRST) SSC Criteria and Capabilities
8. Plant Programs

Additional SAR Content –Outside the Scope of TICAP

9. Control of Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste
10. Control of Occupational Doses
11. Organization and Human-System Considerations
12. Post-construction Inspection, Testing and Analysis Programs

Audit/inspection of Applicant Records

- Calculations
- Analyses
- P&IDs
- System Descriptions
- Design Drawings
- Design Specs
- Procurement Specs
- Probabilistic Risk Assessment

<u>Additional Portions of Application</u>

- Technical Specifications
- Technical Requirements Manual
- Quality Assurance Plan (design)
- Fire Protection Program (design)
- Quality Assurance Plan (construction and operations)
- Emergency Plan
- Physical Security Plan
- Special nuclear material (SNM) physical protection program
- SNM material control and accounting plan
- Cyber Security Plan
- Fire Protection Program (operational)
- Radiation Protection Program
- Offsite Dose Calculation Manual
- Inservice inspection/Inservice testing (ISI/IST) Program
- Environmental Report
- Site Redress Plan
- Exemptions, Departures, and Variances
- FSP (under consideration for Part 53 applications)

- SAR structure based on clean sheet approach

*Additional contents of application outside of SAR are still under discussion. The above list is draft and for illustration purposes only.

ARCAP/TICAP Background

- Status of ARCAP ISG Draft White Papers

ARCAP ISG Title	Date	Accession No.
Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications - Roadmap	Dec 2, 2021	ML21336A702
Chapter 2, "Site Information"	July 6, 2021	ML21189A031
Chapter 9, "Control of Routine Plant Radioactive Effluents, Plant Contamination and Solid Waste	July 6, 2021	ML21189A033
Chapter 10, "Control of Occupational Doses"	July 6, 2021	ML21189A035
Chapter 11, "Organization and Human-System Consideration"	Nov 5, 2021	ML21309A020
Chapter 12, "Post Construction Inspection, Testing and Analysis Program"	Oct 21, 2021	ML21294A266
Licensing Modernization Project-based Approach for Developing Technical Specifications	May 10, 2021	ML21133A490
Risk-Informed, Performance-Based Fire Protection Program (for Operations)	Sept 10, 2021	ML21253A134
Risk-Informed ISI/IST Programs	Aug 4, 2021	ML21216A051

ARCAP/TICAP Background

- Status of TICAP Guidance Documents

TICAP Title	Date	Accession No.
NEI 21-07, Revision 0, Technology Inclusive Guidance for Non-Light Water Reactors Safety Analysis Report Content for Applicants Utilizing NEI 18-04 Methodology	Aug 30, 2021	ML21250A378
RG Draft White Paper, “Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Advanced Reactors”	Dec 2, 2021	ML21336A697

- TICAP guidance documents being revised based on continuing interactions with stakeholders
  - December 14, 2021, public workshop
    - During this meeting, NEI 21-07, Revision 0-B was discussed (ADAMS Accession No. ML21343A292)
  - Planning for another public workshop in mid-January 2022

Advanced Reactor Content of Application Project - Overview

ARCAP – High Level Overview

- ARCAP Roadmap ISG
  - Proposes a 12-chapter SAR guidance structure
  - Guidance for first 8 SAR chapters references the TICAP guidance
  - Chapters 9, 10, 11, and 12 point to individual ISGs developed for each SAR chapter
  - Includes pointers to draft white papers, guidance under development or to be developed for portions of the application outside the SAR
    - Examples of guidance that the staff has developed: TS, Risk Informed ISI/IST, and Fire Protection for Operations
    - Examples of high-level guidance embedded in the Roadmap ISG: Technical Requirements Manual, Quality Assurance Plan, Fire Protection (design), and Offsite Dose Calculation Model
    - Examples of guidance being considered: security, emergency planning, material control and accountability, financial qualification and cyber security

ARCAP – High Level Overview

- ARCAP Roadmap ISG (continued)
  - Includes several appendices:
    - Appendix C on preapplication engagement guidance
      - Based on white paper discussed extensively during advanced reactor public stakeholder meetings
      - Purpose of Appendix C is to capture the white paper guidance in a durable product that will have the benefit of a formal public comment period
    - Appendix D on Analysis of Applicability of NRC Regulations to non-LWRs (plan to include later)
      - Based on white paper discussed extensively during advanced reactor public stakeholder meetings (ADAMS Accession No. ML21175A287)
      - Purpose of Appendix D is to capture the white paper guidance in a durable product that will have the benefit of a formal public comment period

ARCAP – High Level Overview

- ARCAP Roadmap ISG (continued)
  - Includes several appendices (continued):
    - Appendix E on CP guidance
      - Three parts to draft CP guidance
        - Common portion applicable to both LWRs and non-LWRs
          - Will be updated to be consistent with LWR CP ISG when it is issued
        - Portion applicable to LMP based approach – point to TICAP CP guidance
        - Portion applicable to CP guidance outside the scope of TICAP

ARCAP – High Level Overview

- ARCAP Chapter 2: Site Information
  - Supplements information in the SAR that is outside the scope of LMP.
  - Intent is to limit the amount of material in SAR Chapter 2 to what is necessary for establishing safety significant design parameters and for performing the safety analysis, along with its supporting bases.
  - If necessary, additional supporting information (e.g., historical records, geological data, etc.) could be documented in a separate report available for audit.
  - Section 2.6 includes a process for establishing the ground motion response spectrum using the Senior Seismic Hazard Analysis Committee guidance

- ARCAP Chapters 9 and 10: Normal Effluents and Occupational Dose
  - Applies a performance-based approach for level of detail of information provided in the SAR

ARCAP – High Level Overview

- ARCAP Chapter 11: Organization and Human-System Consideration
  - Developed to support near-term Part 50 and 52 applications with a more traditional concept of operations
    - Guidance will be updated later to include concepts discussed as part of the Part 53 proposed rulemaking effort
  - Human Factors Engineering
    - NRC staff identified a need to provide guidance in this area to supplement LMP and the associated pending TICAP guidance
    - LMP provides human factors insights but provides limited guidance on how to develop a HFE program
    - ARCAP Chapter 11 ISG covers HFE information that would support NRC findings

ARCAP – High Level Overview

- ARCAP Chapter 11 – Organization and Human-System Consideration (continued)
  - Operator Licensing
    - Proposed guidance extends beyond what would be expected in an application
      - Centrally located guidance to provide a holistic approach to operator licensing
        - May eventually split out guidance not specifically associated with the content of an application
    - Guidance includes areas such as:
      - Description and qualification of simulator used to administer initial operator licensing examinations
      - Use of simulator for operations training experience and examinations during construction
      - Operator license issuance prior to fuel load
  - Operator Staffing guidance includes areas such as:
    - Option of providing technical basis for control room staffing in conjunction with control room configuration that would support capturing the requirements necessary in a DC rulemaking
    - Provide technical basis that could support a future exemption from §§ 50.54(m) and/or 50.54(k) requirements

ARCAP – High Level Overview

- ARCAP Chapter 12 – Post Construction Inspection, Testing and Analysis Program
  - Intended to provide guidance to the NRC staff regarding application content that would support making the finding that the applicant has met the applicable Part 50 and Part 52 regulations
  - ISG differentiates between 10 CFR Part 52 applicants that must include inspections, tests, analysis and acceptance criteria (ITAAC) and 10 CFR Part 50 applications that are not required to include ITAAC.
  - Requirements to describe preoperational testing and initial operations in OL and COL applications are contained in 50.34(b)(6)(iii) and 52.79(a)(28), respectively.
  - Provides guidance for:
    - post-construction inspection, preoperational testing (i.e., tests conducted following construction and construction-related testing, but prior to initial fuel load), analysis verification, and
    - initial startup testing (i.e., tests conducted during and after initial fuel load, up to and including initial power ascension).

ARCAP – High Level Overview

- ARCAP TS Guidance
  - The text in the 10 CFR 50.36 regulations for TS content require adaptation to correlate to the analysis and outputs of the risk-informed LMP approach described in NEI 18-04.
    - 10 CFR 50.36 requirements for safety limits, limited safety system settings, and limiting condition for operations Criteria 1 through 3 involve challenges to the “integrity of a fission product barrier.”
    - To evaluate the acceptability of risk-informed TS for advanced reactors, this ISG correlates the 10 CFR 50.36 text with appropriate NEI 18-04 process analysis/outputs. These analysis/outputs include:
      - required safety functions
      - SR SSCs
      - frequency-consequence (F-C) target
      - 10 CFR 50.34 dose limits

ARCAP – High Level Overview

- ARCAP Fire Protection for Operations
  - 10 CFR 50.48(a) requires that each operating nuclear power plant have a fire protection plan that meets the requirements of either 10 CFR Part 50, Appendix A, Criterion 3 for LWRs or the applicant's proposed principal design criteria (PDC) that have been deemed acceptable by the NRC.
    - Although 10 CFR 50.48(c) – NFPA 805 – does not apply to non-LWRs concepts associated with this risk-informed approach are included in the draft ISG
  - The scope of this ISG addresses the review of the application content regarding the fire protection program for operations including application descriptions of:
    - Management policy and program direction and the responsibilities of those individuals responsible for the program/plan's implementation.
    - The integrated combination of procedures and personnel that will implement fire protection program activities.

ARCAP – High Level Overview

- ARCAP Risk-Informed ISI/IST
  - ISG purpose is to facilitate the review of advanced reactor applications that use a risk-informed approach to developing their ISI/IST programs
  - ISG guidance requires the use of risk information from a plant-specific PRA that is in conformance with an NRC endorsed PRA standard
  - For advanced LWRs, guidance on how to risk-inform ISI/IST already exists (RG 1.175, and 1.178) and is used in the ISG
  - 10 CFR Part 50 contains only general requirements (e.g., 50.34(b)(6)(iv)) related to non-LWR ISI/IST programs, although ASME has recently issued Section XI, Division 2 - NRC has reviewed and issued DG-1383 for public comment on endorsement of ASME Section XI, Division 2
  - For non-LWR ISI, applicants are expected to use the risk information from their plant-specific PRA to identify the piping, reactor coolant boundary, pressure retaining and passive components and their supports to be included in the program, along with other components whose failure could prevent a safety function from being accomplished

ARCAP – High Level Overview

- ARCAP Risk-Informed ISI/IST (continued)
  - For non-LWR IST, applicants are expected to use the risk information from their plant-specific PRA and associated design reviews to identify the active valves, pumps and dynamic restraint devices and the passive components with active safety functions to be included in the program.
  - For non-LWR ISI, the ISG is based upon the applicant using the requirements in ASME Section XI, Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants” (which is the subject of draft RG-1383).

Technology Inclusive Content of Application Project - Overview

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

TICAP – High Level Overview

- TICAP guidance
  - Goal is to develop technology-inclusive guidance that proposes an optional formulation of advanced reactor application content that is based on a risk-informed, performance-based approach for demonstrating that plant safety meets the underlying intent of the current requirements
  - Guidance is intended to increase efficiency of developing and reviewing an application
  - Scope is governed by the LMP process to facilitate a systematic, technically acceptable, and predictable approach for developing key portions of a design's SAR
    - LMP provides process for identifying LBEs, determining SSC categorization, establishing special treatments for SSCs, and ensuring DID adequacy

TICAP – High Level Overview

- TICAP SAR Structure
  - Chapter 1 - General Plant and Site Description, And Overview of The Safety Case
    - The information in this chapter should allow the reviewer to obtain a basic understanding of the overall facility, such as the type of permit, license, certification or approval requested, the number of plant units, a brief description of the proposed plant location, and the type of advanced reactor being proposed.
  - Chapter 2 - Methodologies and Analyses
    - An important part of the design process for reactor designs is the identification of events that could challenge key safety functions and layers of defense against the release of radioactive materials. Therefore, a key part of the review of an advanced reactor application is the selection of LBEs.

TICAP – High Level Overview

- TICAP SAR Structure
  - Chapter 3 – Licensing Basis Events
    - The information in this chapter should describe the systematic and reproducible process and methodology used to select the LBEs, and the specific analysis and evaluation of the selected LBEs for the proposed design. The analysis in this section is primarily described in terms of event sequences comprised of an initiating event, the plant response to the initiating event (which includes a sequence of successes and failures of mitigating systems) and a well-defined end state. The consequences from LBEs are expressed as dose at the exclusionary boundary and compared to the F-C curve in LMP documentation.
  - Chapter 4 – Integrated Evaluations
    - The information in this chapter should describe the integrated risk of all LBEs selected for the proposed design and evaluated against three cumulative risk targets and the DID evaluation.

TICAP – High Level Overview

- TICAP SAR Structure
  - Chapter 5 - Safety Functions, Design Criteria, and Systems, Structures, and Components Classification
    - As part of the LMP process, LBEs are generally defined in terms of successes and failures of SSCs that perform safety functions and are modeled in the PRA. Therefore, the PRA safety functions are those functions responsible for the prevention and mitigation of an unplanned radiological release from any source within the plant.
    - The information in this chapter should describe the proposed PDC necessary to ensure that the "important to safety" SSCs provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public
    - The information in this chapter should describe the approach for designating SSC classifications

TICAP – High Level Overview

- TICAP SAR Structure
  - Chapter 6 - Safety-Related Systems, Structures, and Components Criteria and Capabilities
    - The information in this chapter should leverage the analysis performed for the SR SSCs in Chapter 5 of NEI 21-07 and describe in further detail the criteria, capabilities and special treatment of all SR SSCs.
  - Chapter 7 – Non-Safety-Related Special Treatment (NSRST) Systems, Structures, and Components Criteria and Capabilities
    - The information in this chapter should describe the regulatory design and special treatment requirements for SSCs classified as NSRST. NSRST SSCs are relied upon to perform risk-significant functions. Risk-significant SSCs are those that perform functions that prevent or mitigate any LBE from exceeding the F-C target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs. SSCs considered necessary for ensuring DID adequacy are also categorized as NSRST.

TICAP – High Level Overview

- TICAP SAR Structure
  - Chapter 8 – Plant Programs
    - The information in this chapter should provide information on those plant programs relied upon to provide special treatment to SR and NSRST SSCs that are part of the affirmative safety case. The information should provide an overview of the special treatment programs, addressing the purpose, scope, and performance objectives as well as applicability to SSCs. The information for the programs should provide reasonable assurance that 1) reliability and performance targets are met, and 2) safety-significant uncertainties are addressed. Program areas could include human factors, quality assurance, startup testing, and equipment qualification, among others.

TICAP – High Level Overview

- TICAP - PDC
  - Development of an LMP-based approach for developing proposed PDCs has been a subject of ongoing stakeholder interactions with the most recent occurring at the 12/14 TICAP public workshop
  - Issues discussed include:
    - Applicability of General Design Criteria (GDC) in 10 CFR part 50, Appendix A, including their scope, to non-LWR advanced reactor applicants
    - Additional guidance available to advanced reactor applicants for developing proposed PDC
    - The possible need for exemptions to applicable regulations for proposed PDC developed using the LMP methodology

TICAP – High Level Overview

- Summary of TICAP – PDC Discussions
  - PDC are required to be proposed by applicants for the following:
    - ✓ 10 CFR 50.34(a)(3) for CPs
    - ✓ 10 CFR 52.79(a)(4) for COLs
    - ✓ 10 CFR 52.47(a)(3) for DCs
    - ✓ 10 CFR 52.137(a)(3) for SDAs
    - ✓ 10 CFR 52.157(a) for MLs
  - PDC are a means to meet the requirements of the AEA, Section 182 for inclusion in license applications of *'the specific characteristics of the facility, and such other information as the Commission may, by rule, or regulation, deem necessary in order to enable it to find that the utilization or production of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public.'*

TICAP – High Level Overview

- Summary of TICAP – PDC Discussions (continued)
  - GDC are applicable to LWRs (“minimum requirements”) and “**provide guidance** to applicants for construction permits in establishing principal design criteria **for other types of nuclear power units.**”
  - Advanced Reactor Design Criteria (ARDC) developed by the NRC in RG 1.232 are intended to provide insight into the staff’s views on how the underlying safety bases for the GDC could be applied to address non-LWR design features. As noted in RG 1.232, the development of the ARDC was an important first step to address the unique characteristics of non-LWR technology but the NRC recognizes the future benefits to risk-informing the non-LWR design criteria and determining the role of such criteria within a new regulatory framework.

TICAP – High Level Overview

- Summary of TICAP – PDC Discussions (continued)
  - The NRC position on the requirement for proposed PDC is that it includes the scope of PDC described in the regulations as well as in the regulatory and judicial history.

    The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

  - PDC are particularly important for CP applications since CP applicants are required to provide less information, comparatively speaking, and the information that is provided is preliminary.
  - Proposed PDC play a significant role in supporting the NRC's finding that there is reasonable assurance that safety questions will be satisfactorily resolved and that the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

TICAP – High Level Overview

- Summary of TICAP – PDC Discussions (continued)
  - Proposed PDC determined to be necessary for a non-LWR design and submitted in an application under 10 CFR Part 50 or Part 52 should be as comprehensive in scope as the GDC and ARDC (i.e., establish the necessary design, fabrication, construction, testing, and performance requirements).
  - Non-LWR applicants proposing PDC that do not fully address the scope of PDC (i.e., design, fabrication, construction, testing, and performance requirements) will need to request exemptions from the applicable regulations.
  - Non-LWR applicants must provide supporting information that justifies to the NRC how their design meets their proposed PDC and how their proposed PDC demonstrate reasonable assurance of safety.
  - NRC believes that it is feasible for applicants for CPs, COLs, DCs, SDAs and MLs to provide justification for an exemption by ensuring that the elements of the PDC scope not fully addressed in their proposed PDC are included in their application.

TICAP – High Level Overview

- Summary of TICAP – PDC Discussions (continued)
  - TICAP guidance document NEI 21-07 proposed an approach to supplement PDC focused on SR SSCs with proposed Complementary Design Criteria (CDC) that focus on NSRST SSCs.
  - In "fitting" the LMP approach to developing PDC and CDC into the Part 50 and Part 52 regulatory framework, the NRC concluded that both PDC and CDC would need to be relied on the NRC to make its regulatory finding.
  - NRC suggested that a two-tiered PDC approach would comply with the regulations (i.e., PDC Type A for functions performed by SR SSCs and PDC Type B for functions performed by NSRST SSCs).
  - NRC expects further stakeholder interactions at the next TICAP public workshop in January 2022.

Next Steps – Future Milestones

TICAP Near-Term Milestones	Target Date
Update of NRC Draft Guidance Documents	Early December 2021
Continuation of Discussion of NRC draft Exceptions, Clarifications, and Additions (possibility of future draft industry or staff documents)	TBD
NEI 21-07, Revision 1	TBD
Issuance of TICAP draft RG and ARCAP ISG for public comment	Early Calendar Year 2022

Discussion

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MEETING BREAK

Meeting to resume in 1 hour

Nuclear Energy Institute / U.S. Nuclear Industry Council Presentation

Final Discussion and Questions

Acronyms and Abbreviations

ACRS	Advisory Committee for Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act
ANS	American Nuclear Society
AOO	Anticipated operational occurrence
ARCAP	Advanced Reactor Content of Application Project
ARDC	Advanced reactor design criteria
ARPS	Advanced reactor policy statement ("Policy Statement on the Regulation of Advanced Reactors;" 73 FR 60612; October 14, 2008)
ASME	American Society of Mechanical Engineers
BDBA	Beyond design basis accident
BDBE	Beyond design basis event
CCFP	Conditional containment failure probability
CDC	Complementary design criteria
CDF	Core damage frequency
CFR	Code of Federal Regulations
COL	Combined license
CP	Construction permit

DANU	Division of Advanced Reactors and Non-Power Production and Utilization Facilities
DBA	Design basis accident
DBE	Design basis event
DC	Design certification
DG	Draft regulatory guide
DID	Defense-in-depth
D-RAP	Design reliability assurance program
EP	Emergency preparedness
EPRI	Electric Power Research Institute
ESP	Early site permit
F-C	Frequency-consequence
FMEA	Failure modes and effects analysis
FR	*Federal Register*
FRN	*Federal Register* Notice
FSAR	Final safety analysis report
FSP	Facility safety program
FTA	Fault tree analysis
GDC	General design criteria

Acronyms and Abbreviations

HAZOP	Hazard and operability
HFE	Human factors engineering
HRA	Human reliability analysis
HSI	Human-system interface
HTGR	High temperature gas cooled reactor
IAEA	International Atomic Energy Agency
IEFR	Individual early fatality risk
ILCFR	Individual latent cancer fatality risk
ISG	Interim staff guidance
ISI	Inservice inspection
IST	Inservice testing
ITAAC	Inspections, tests, and acceptance criteria
LB	Licensing basis
LBE	Licensing basis event
LERF	Large early release frequency
LMP	Licensing Modernization Project
LMR	Liquid metal cooled reactor
LRF	Large release frequency

LWA	Limited work authorization
LWR	Light water reactor
MCA	Maximum credible accident
MHA	Maximum hypothetical accident
ML	Manufacturing license
MLD	Master logic diagram
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NGO	Non-governmental organization
non-LWR	Non-light water reactor
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSRST	Non-safety-related special treatment
NUREG	U.S. NRC technical report designation
NWPA	Nuclear Waste Policy Act
OFR	Office of the Federal Register
OL	Operating license
OMB	Office of Management and Budget

Acronyms and Abbreviations

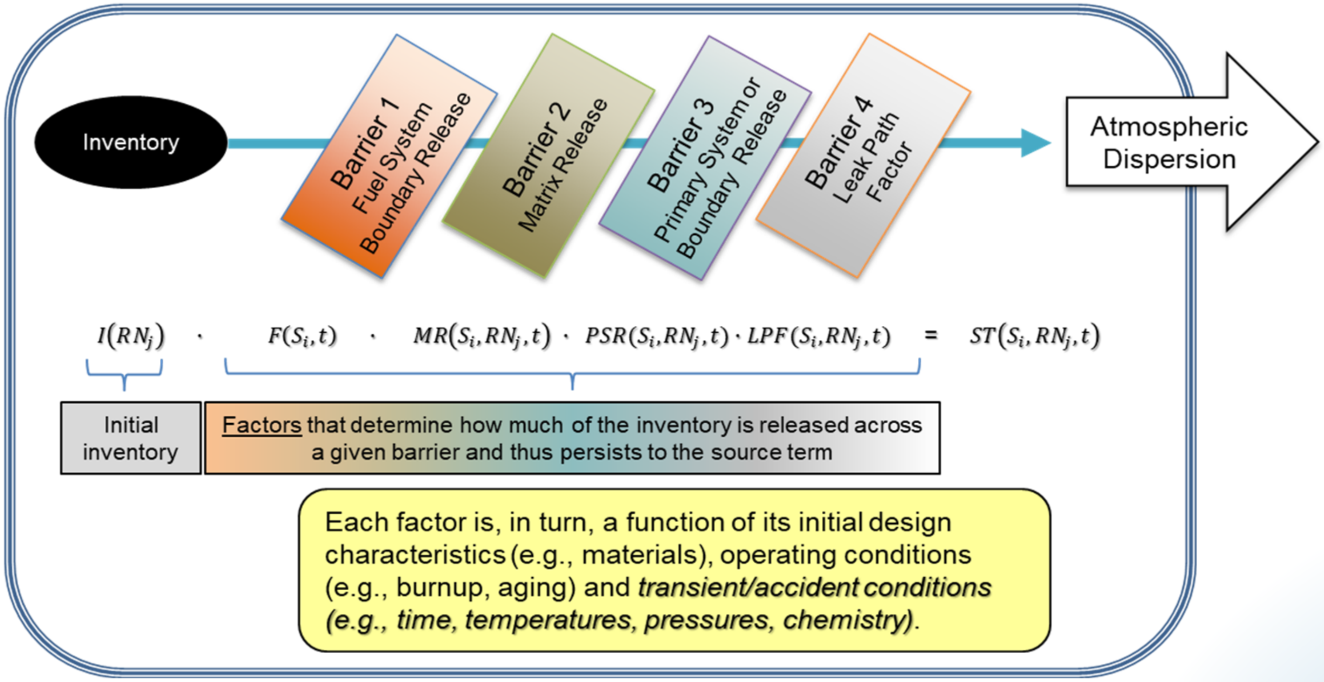
ONT	Other new technology
P&ID	Piping and instrumentation diagrams
PDC	Principal design criteria
PRA	Probabilistic risk assessment
PRAPS	PRA policy statement ("Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities;" 60 FR 42622; August 16, 1995)
PSA	Probabilistic safety assessment
QHO	Quantitative health objective
RG	Regulatory guide
RIM	Reliability and Integrity Management
RIN	Regulation identifier number
SAMDA	Severe accident mitigation design alternative
SAPS	Severe accident policy statement ("Severe Reactor Accidents Regarding Future Designs and Existing Plants;" 50 FR 32138; August 8, 1985)
SAR	Safety analysis report
SAT	Systems approach to training
SDA	Standard design approval

SECY	Office of the Secretary
SGPS	Safety goal policy statement ("Safety Goals for the Operation of Nuclear Power Plants;" 51 FR 28044; August 4, 1986; as corrected and republished at 51 FR 30028; August 21, 1986)
SMR	Small modular reactor
SNM	Special nuclear material
SR	Supporting requirement (NLWR PRA standard)
SR	Safety-related
SRM	Staff requirements memorandum
SSC	Systems, structures, and components
SSR	Specific safety requirement (IAEA)
STA	Shift technical advisor
TICAP	Technology Inclusive Content of Application Project
TIRIMA	Technology-inclusive, risk-informed, maximum accident
TMI	Three Mile Island
TR	Technical report
TS	Technical specifications

Background Slides

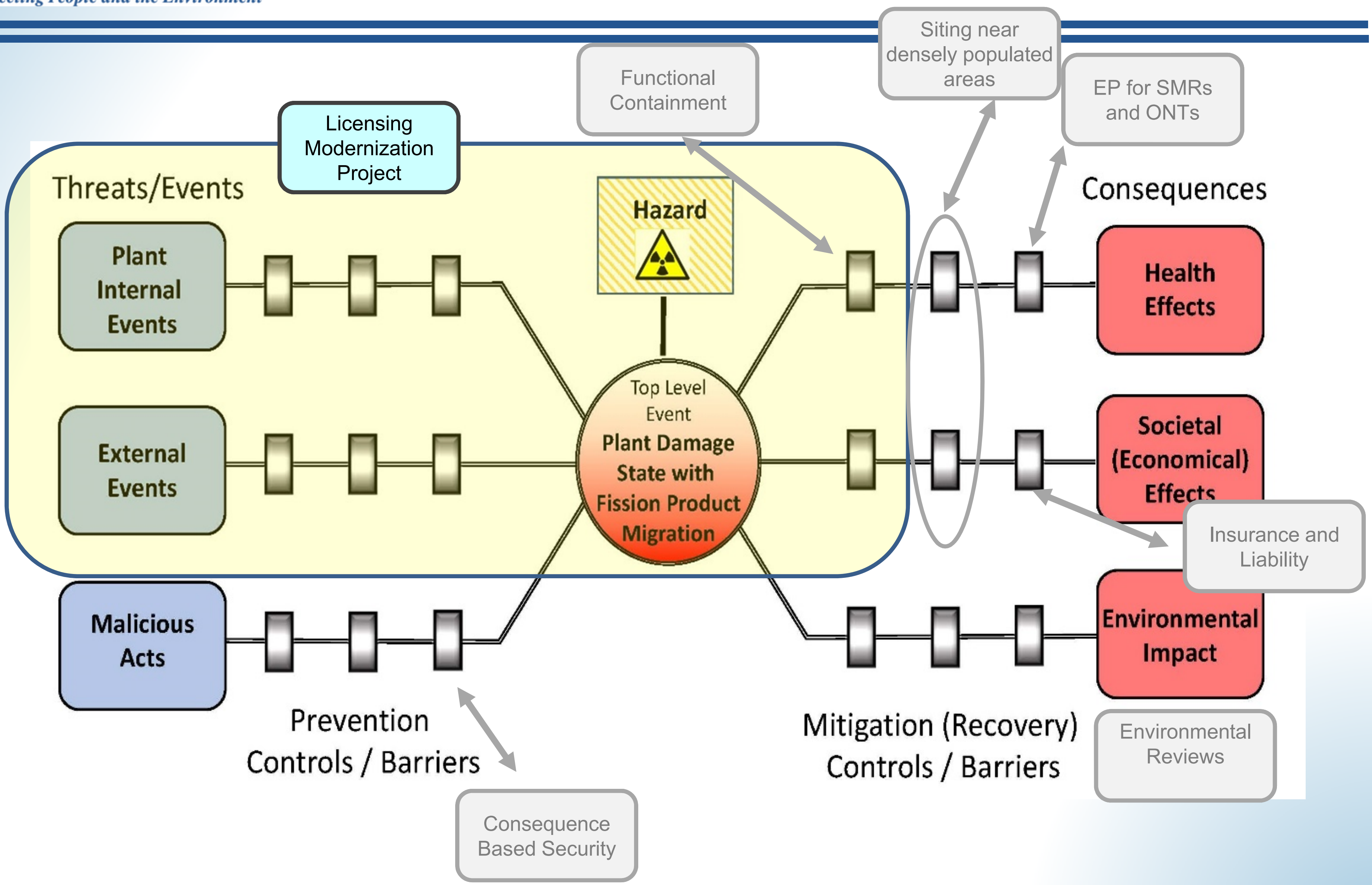
First Principles

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



See: SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors," and INL/EXT-20-58717, "Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities"

Integrated Approach



Background

- Nuclear Energy Innovation and Modernization Act (NEIMA; Public Law 115-439) signed into law in January 2019 requires the NRC to complete a rulemaking to establish a technology-inclusive, regulatory framework for optional use for commercial advanced nuclear reactors no later than December 2027
  - (1) ADVANCED NUCLEAR REACTOR—The term “advanced nuclear reactor” means a nuclear fission or fusion reactor, including a prototype plant… with significant improvements compared to commercial nuclear reactors under construction as of the date of enactment of this Act, …

Severe Accidents

- Severe Accident Policy Statement
  - Although in the licensing of existing plants the Commission has determined that these plants pose no undue risk to public health and safety, this should not be viewed as implying a Commission policy that safety improvements in new plant designs should not be actively sought. The Commission fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs.
- 10 CFR 52.47(a)(23)
  - For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass
- NUREG-1226 (Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants; Executive Summary)
  - (4) While the Final Policy Statement encourages innovative reactor designs and safety criteria, the review of advanced reactor designs will still require satisfactory consideration of the Commission's regulations, regulatory guides and other guidelines, such established and developing criteria as the defense-in-depth philosophy, standardization, the Commission's safety goal and severe accident policies, and applicable industry codes and standards.

Backup Slides

Technology-Inclusive Content of Application (TICAP) and Advanced Reactors Content of Application (ARCAP)- Nexus to Part 53

Part 53- Proposed Structure

- A- General Provisions
- B- Tech-Incl Safety Requirements
- C- Design and Analysis Req.
- D- Siting
- E- Const. and Manufacturing
- F- Operations
- G- Decommissioning
- H- Licenses, Cert, and Approvals
- I- Maintaining/Revising LB Info
- J- Administrative requirements

Subpart H – Licenses, Cert, and Approvals

This subpart is envisioned to address requirements for initial applications for licenses, certifications, or approvals. The subpart will support the licensing processes available under the Part 50 and Part 52 frameworks. Assessment and update of manufacturing licenses is possible.

ARCAP-Guidance for Content of Application Guidance (Roadmap)

TICAP - LMP-based portions of FSAR that are related to:

- LBEs
- Safety classification
- Defense in Depth

ARCAP specific chapter guidance – examples:

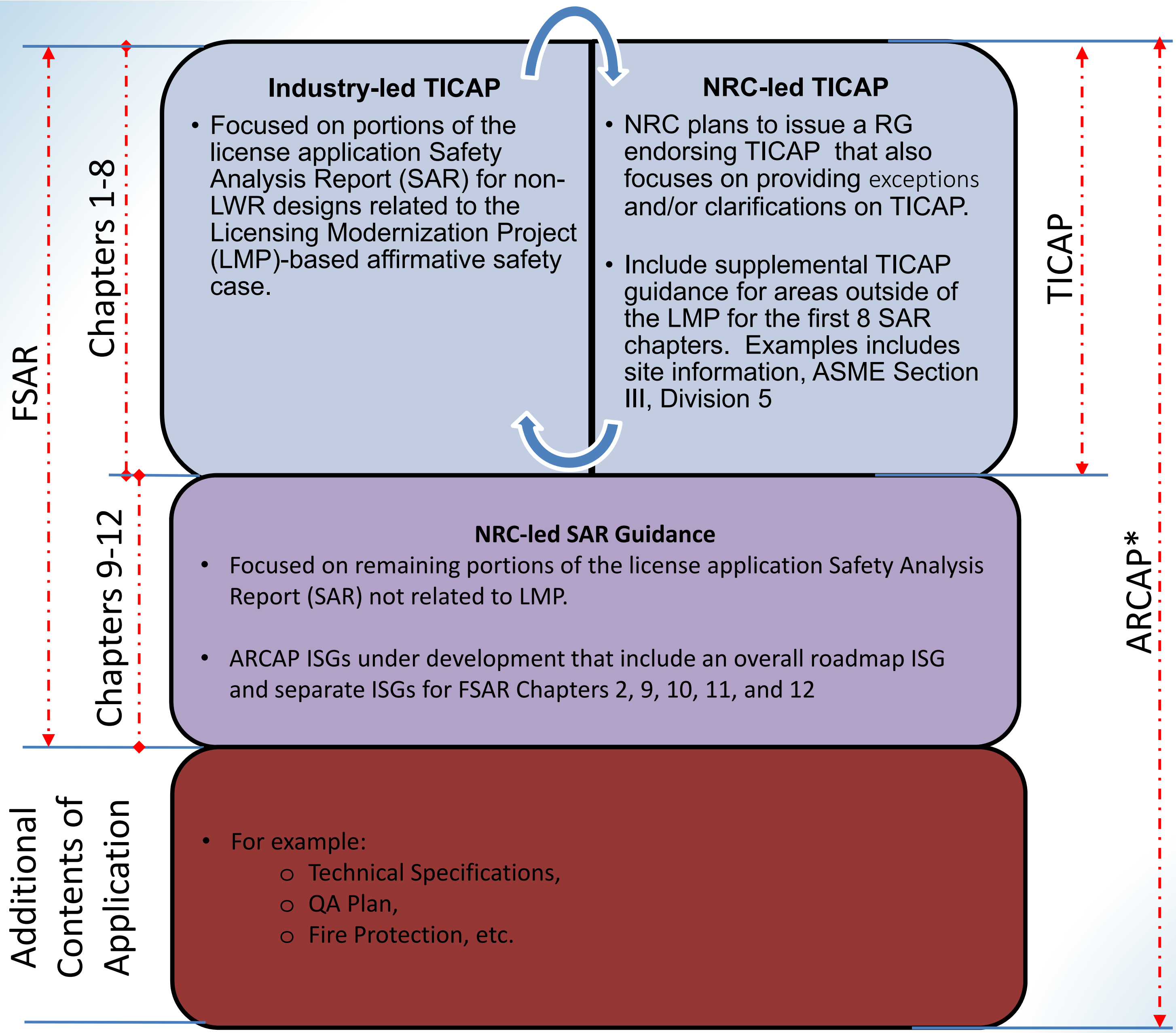
- Site information
- ARCAP chapters 9, 10, 11, and 12

Guidance under Development examples:

- Fuel Qualification
- ASME Section III, Division 5
- EP rulemaking
- Security Rulemaking

Existing Regulatory Guidance

Note: The illustrated content structure for Part 53 (including Subpart H) is part of ongoing work and subject to change.



*Staff plans to issue an ARCAP Roadmap ISG that would provide pointers to various guidance documents developed/issued.

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Key Part 53 Guidance by Subpart

<table>
<tr><th colspan="2">Subpart A: General Provisions</th></tr>
<tr><td><u>Existing / Ongoing Guidance</u></td><td><u>Additional Guidance</u></td></tr>
<tr><td colspan="2">N/A</td></tr>
<tr><th colspan="2">Subpart B: Safety Criteria</th></tr>
<tr><td><u>Existing / Ongoing Guidance</u></td><td><u>Additional Guidance</u></td></tr>
<tr><td>N/A</td><td>▪ Further explanation of criteria and structure in the Statements of Consideration</td></tr>
<tr><th colspan="2">Subpart C: Design and Analysis</th></tr>
<tr><td><u>Existing / Ongoing Guidance</u></td><td><u>Additional Guidance</u></td></tr>
<tr><td>▪ NEI 18-04 & RG 1.233 (LMP)
▪ ANS/ASME-RA-S-1.4 (Non-LWR PRA Standard)
▪ Industry PRA Peer Review Guidance for Non-LWRs (NEI 20-09)
▪ ANS/ASME Standards (ASME III-5, ASME XI-II)
▪ Fuel Qualification (NUREG-2246)
▪ RG 1.232 (ARDCs)</td><td>▪ ISG on PRA for Initial Licensing
▪ RG 1.247 Endorsing Non-LWR PRA Standard and NEI Peer Review Guidance
▪ Application of Analytical Margins
▪ Treatment of Chemical Hazards</td></tr>
<tr><th colspan="2">Subpart D: Siting Requirements</th></tr>
<tr><td><u>Existing / Ongoing Guidance</u></td><td><u>Additional Guidance</u></td></tr>
<tr><td>▪ SECY-20-0045/RG 4.7
▪ External Hazard Updates
▪ Risk-Informed Seismic Design; ANS 2.26</td><td>N/A</td></tr>
</table>

Key Part 53 Guidance by Subpart

Subpart E: Construction and Manufacturing	
Existing / Ongoing Guidance	Additional Guidance
N/A	▪ Manufacturing Guidance ▪ QA Alternatives
Subpart F: Operations	
SSCs	
Existing / Ongoing Guidance	Additional Guidance
▪ NEI 18-04 & RG 1.233 (LMP)	▪ Technical Specifications ▪ Special Treatment ▪ Maintenance, Repair & Inspection ▪ Facility Safety Program
Personnel	
Existing / Ongoing Guidance	Additional Guidance
▪ DRO Paper/preliminary ISG	▪ Concept of Operations
Programs	
Existing / Ongoing Guidance	Additional Guidance
▪ EPZ Draft Final Rule, RG 1.242 ▪ Radiation Protection (ARCAP)	▪ Emergency Preparedness ▪ Security Programs (e.g., FFD, Access Authorization, Cyber Security) ▪ Integrity Assessment Program

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Key Part 53 Guidance by Subpart

Subpart G: Decommissioning	
Existing / Ongoing Guidance	Additional Guidance
N/A	N/A
Subpart H: Licensing	
Existing / Ongoing Guidance	Additional Guidance
▪ TICAP ▪ ARCAP	▪ Manufacturing Licenses
Subpart I: Maintaining Licensing Basis	
Existing / Ongoing Guidance	Additional Guidance
N/A	▪ 50.59 Equivalent ▪ FSAR/PRA Updates
Subpart J: Administrative/Misc.	
Existing / Ongoing Guidance	Additional Guidance
N/A	▪ Reporting Requirements ▪ Financial/Liability