

# **10 CFR Part 53**

## **“Licensing and Regulation of Commercial Nuclear Plants”**

Discussion of Selected Part 53 Topics

March 29, 2022

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# Agenda

- 1:00pm – 1:15pm Welcome / Introductions / Logistics / Goals
- 1:15pm – 3:45pm Discussion of Selected Part 53 Topics
- QHOs as a performance metric
  - Beyond Design Basis Events
  - ALARA requirements
  - Facility Safety Program as an operation program
  - Additional topics
- 3:45pm – 4:00pm Questions and Closing Remarks

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# Welcome & Introductions

## Welcome:

- Rob Taylor, Office of Nuclear Reactor Regulation (NRR)

## NRC Speakers/Presenters:

- Bob Beall, Office of Nuclear Material Safety and Safeguards – Rulemaking PM & Meeting Facilitator
- Nanette Valliere, NRR – Technical Lead
- Bill Reckley, NRR

## Public Meeting Slides: ADAMS Accession No.

- ML22082A022

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# Purpose of Today's Meeting

- Discuss Part 53 proposed rulemaking effort.
- Today's meeting is a "Comment-Gathering" meeting, which means that public participation is actively sought in the discussion of the regulatory issues during the meeting.
- The meeting is being transcribed and the transcription will be available with the meeting summary by April 29, 2022.
- No regulatory decisions will be made at today's meeting.

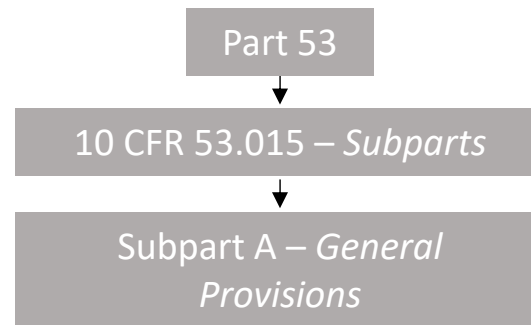
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# General Approach

- NRC staff preparing a proposed rule for Commission consideration and eventual issuance for public comment
- Interactions on preliminary proposed rule text are helpful but should not be confused with the notice and comment process required for a proposed rule

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- Primary focus of today's discussions is on the preliminary proposed rule language for Framework A
  - Future public meetings will be held to discuss Framework B

# Part 53 Licensing Frameworks



## Framework A

- Subpart B – Safety Requirements
- Subpart C – Design Requirements
- Subpart D – Siting
- Subpart E – Construction
- Subpart F – Operations
- Subpart G – Decommissioning
- Subpart H – Licensing Processes
- Subpart I – License Maintenance
- Subpart J – Reporting
- Subpart K – Quality Assurance

## Framework B

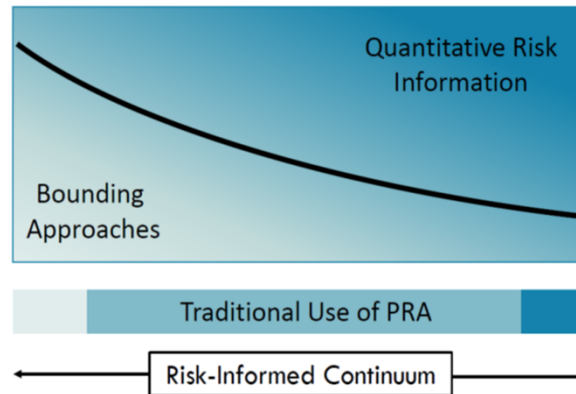
- Subpart N – Purpose/Definitions
- Subpart O – Construction
- Subpart P – Operations
- Subpart Q – Decommissioning
- Subpart R – Licensing Processes
- Subpart S – License Maintenance
- Subpart T – Reporting
- Subpart U – Quality Assurance

Alternate  
Evaluation  
for Risk  
Insights



# Regulatory Options (Frameworks)

Framework B  
Emphasis  
Design Criteria



Framework A  
Emphasis  
Risk metrics and insights

Plant design envelope

Operational states		Accident conditions	
NO	AOO	DBAs	Design Extension Conditions
			Without significant fuel degradation
			With core melting (severe accidents)
Loads and conditions generated by External & Internal Hazards (for each plant state)			
Criteria for functionality, capability, margins, layout and reliability (for each plant state)			
Design basis of equipment for Operational states	Design Basis of Safety Systems including SSCs necessary to control DBAs and some AOOs	Design Basis of safety features for DECs including SSCs necessary to control DECs	
		Features to prevent core melt	Features to mitigate core melt (Containment systems)

FIG. 2. Main elements of the design basis of SSCs for different plant states.

- Traditional approach represented by figure from IAEA guidance.

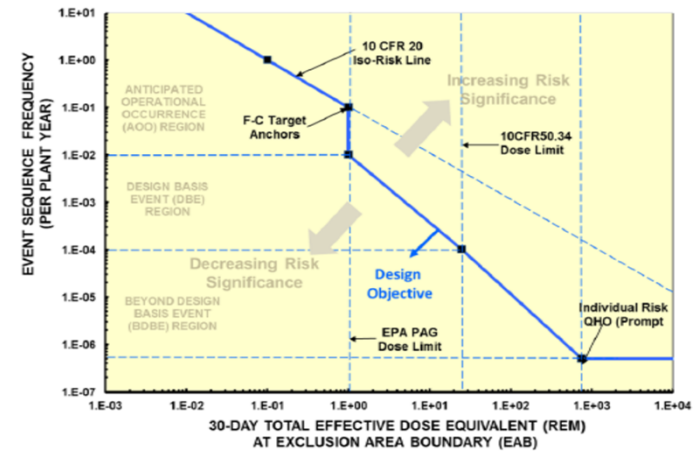


Figure 3-1. Frequency-Consequence Target

- With addition of DBA used to set design criteria and performance objectives for the design of Safety Related SSCs.

# Part 53 Issues

Topic	Addressed in Preliminary Proposed Rule Language
Duplicative/overlapping programs	<ul style="list-style-type: none"> <li>• Quality Assurance requirements consolidated in Subpart K.</li> <li>• Added flexibility for licensees to organize and combine programs, as appropriate, to avoid duplication (Subparts F &amp; K).</li> </ul>
Manufacturing license (ML) expansion	Expanded activities permitted under ML to include fabrication of entire reactor, including fuel loading (Subparts E & H).
Safety criteria structure	Eliminated two-tiered approach to safety criteria (Subpart B).
Codes and standards	Enabled flexibility in using codes and standards.
Normal operations	Decoupled requirements for normal operation from those for licensing basis events (Subparts B & C).
Use of “advanced nuclear plant” and expansion beyond commercial reactors	<ul style="list-style-type: none"> <li>• The staff has removed references to “advanced nuclear plant”.</li> <li>• No plans to expand applicability to research and test reactors (note that NEIMA is directed at commercial reactors) (Subpart A).</li> </ul>



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# Part 53 Issues

<b>Topic</b>	<b>Considerations</b>
Include risk-informed licensing approaches beyond Licensing Modernization Project (LMP)	NRC staff is preparing alternatives to probabilistic risk assessment (PRA)-led analyses (Framework A), including traditional uses of PRA and limited PRA options (Framework B, formerly Part 5X).
Quantitative Health Objectives (QHOs)	Presented on subsequent slides.
Beyond Design Basis Events (BDBEs)	Presented on subsequent slides.
As Low As Reasonably Achievable (ALARA) considerations in design	Presented on subsequent slides.
Facility Safety Program (FSP)	Presented on subsequent slides.

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# Stakeholder Presentations and Discussions

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# Subpart B – QHOs

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## QHOs – Updated Preliminary Proposed Rule Language (March 2022)

§ 53.220 Safety criteria for licensing basis events other than design basis accidents.

Design features and programmatic controls must be provided to:

(a) Ensure plant structures, systems and components (SSCs), personnel, and programs provide the necessary capabilities and maintain the necessary reliability to address licensing basis events in accordance with § 53.240 and provide measures for defense-in-depth in accordance with § 53.250; and

(b) Maintain overall cumulative plant risk from licensing basis events other than design basis accidents analyzed in accordance with § 53.450(e) such that the calculated risk to an average individual within the vicinity of the plant receiving a radiation dose with the potential for immediate life-threatening health effects remains below five in 10 million years, and the calculated risk to such an individual receiving a radiation dose with the potential to cause latent life-threatening health effects remains below two in one million years.

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# QHOs – Basis

- Performance-based approaches use measurable or calculable performance metrics.
- QHOs are well established and have been used in making regulatory decisions since they were developed as part of the NRC’s Safety Goal Policy Statement. Examples include:
  - Regulatory Guide (RG) 1.174 (Using PRA in risk-informed decisions - licensing basis)
  - NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission.”
- Supports risk-informed, performance-based approach as encouraged by NEIMA.
- Provides predictability and stability in that acceptance criteria are defined and are used by both applicants and NRC during initial licensing reviews and maintenance of licensing basis information (Subpart I, “Maintaining and Revising Licensing Basis Information”).

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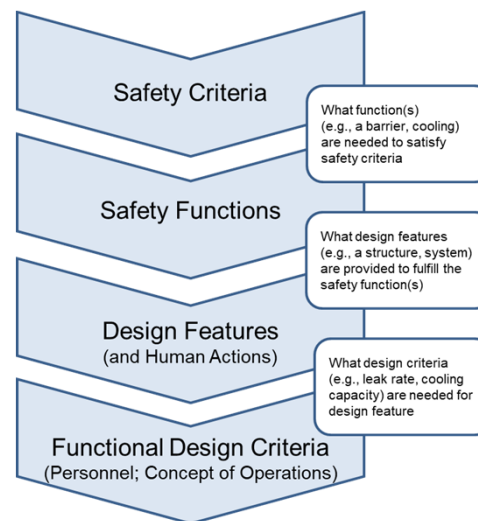
# QHOs – Basis

- Methodologies available for performing risk assessments and comparing to QHOs.
  - Supported by recently issued RG 1.247, “TRIAL - Acceptability of Probabilistic Risk Assessment Results for Non-Light Water Reactor Risk-Informed Activities.”
- Applicants may choose to use surrogate measures to show that designs or plants satisfy the QHO-related criteria (e.g., core damage frequency for light-water reactors).
- Recent language change to “calculated risk” and to refer to “analyzed in accordance with § 53.450(e)” intended to address issues about uncertainties associated with estimating risks to the public from the release of radionuclides.
- Rationale for using QHOs as a metric will be provided in Statement of Considerations for the proposed rule package.
- Added “life-threatening” to maintain alignment with Safety Goal Policy Statement.

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# QHOs – Questions for Stakeholders

- What is a proposed alternative performance metric for Framework A?
- How would an alternative performance metric for Framework A provide sufficient clarity to support applications and associated NRC reviews for a variety of reactor technologies? Is there an issue with the top-down approach included in Framework A (see below)?



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# Stakeholder Presentations and Discussions



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# Subparts B & C – Beyond Design Basis Events

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# BDBEs – Updated Preliminary Proposed Rule Language

**§ 53.210 Safety criteria for design basis accidents.**

**§ 53.220 Safety criteria for licensing basis events other than design basis accidents.**

**§ 53.240 Licensing basis events.**

**§ 53.450(e) Analysis of licensing basis events other than design basis accidents.**

- Anticipated Operational Occurrences (LMP – AOOs)
- Unlikely Event Sequences (LMP – DBEs)
- Very Unlikely Event Sequences (LMP – BDBEs)

❖ Note that Part 53 terminology developed to avoid potential conflicts with Parts 50/52 with terms such as DBE and BDBE while maintaining an alignment with NEI 18-04 and RG 1.233. Other terms such as “design basis” and “important to safety” not used within Framework A for similar reasons.

# Part 53 – Framework A

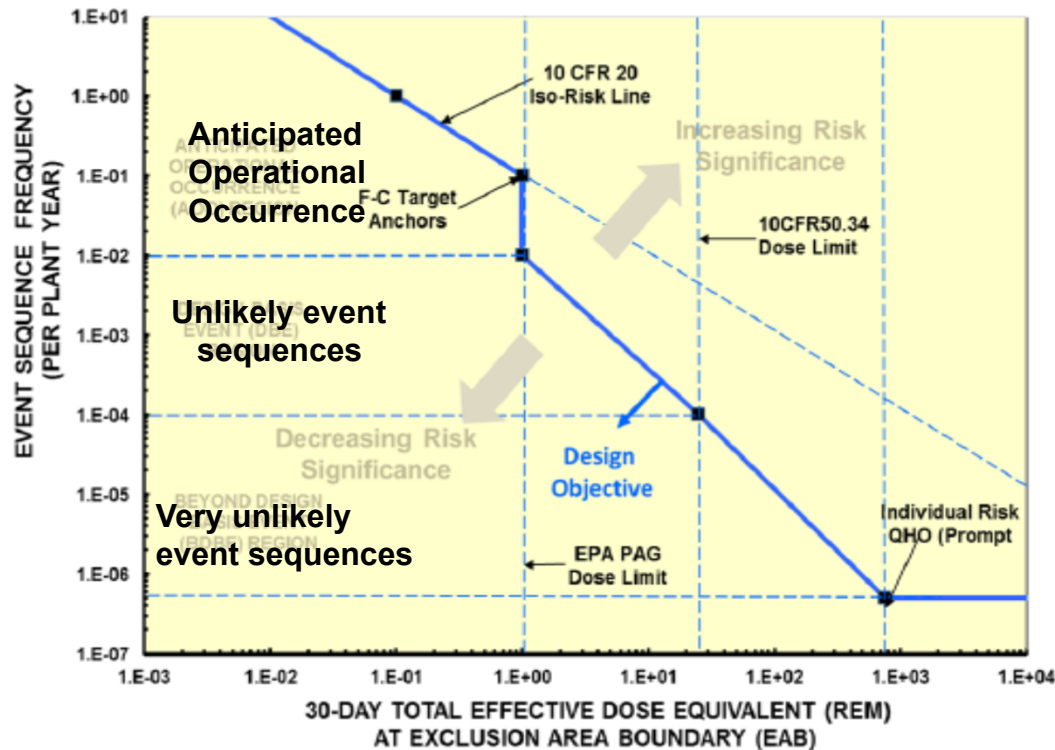


Figure 3-1. Frequency-Consequence Target

*Example of licensing basis events other than design basis accidents with related evaluation criteria*

*Design basis accident (DBA) included within licensing basis events to establish the functional design criteria for Safety Related SSCs*

*Inclusion of very unlikely event sequences integral to addressing the risks posed by a proposed commercial nuclear power plant under Framework A*

*Measures to address very unlikely event sequences primarily addressed through NSRSS design features and associated programmatic controls*

# Traditional Approach (IAEA)

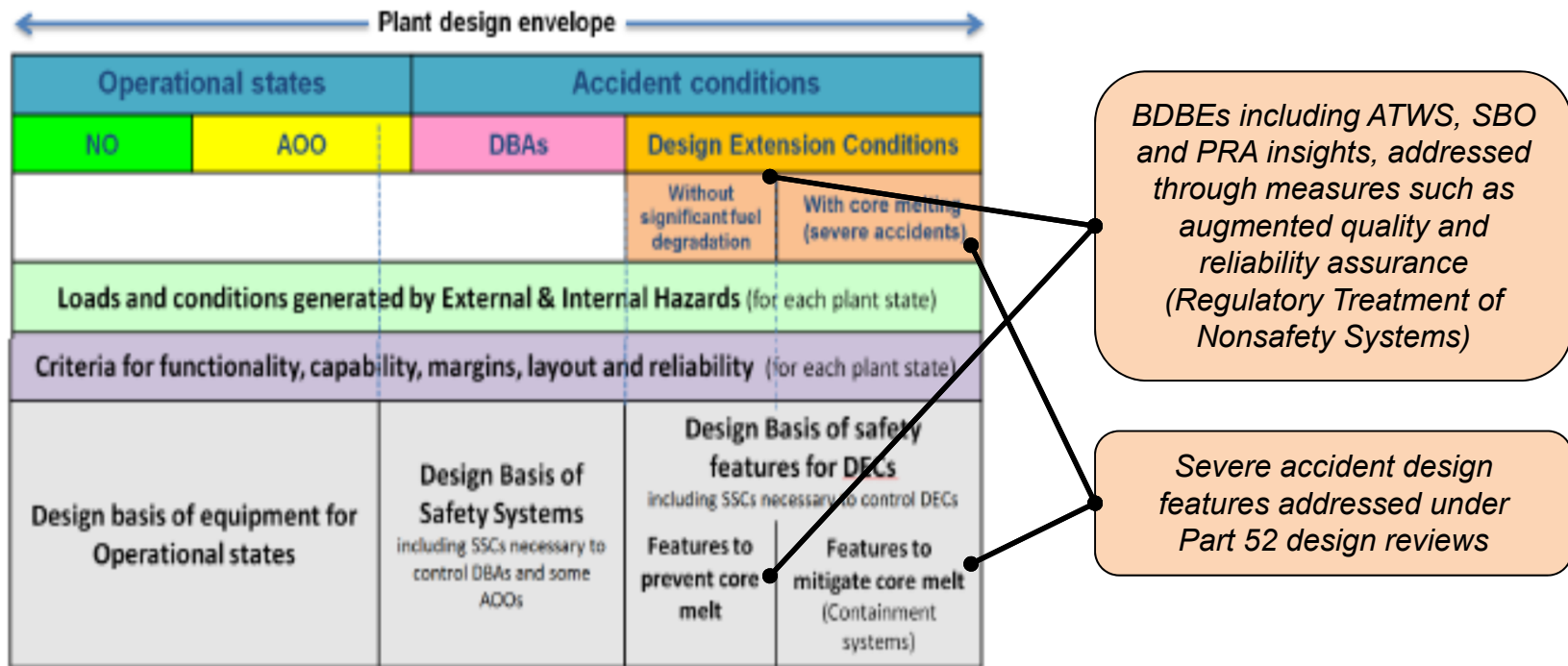


FIG. 2. Main elements of the design basis of SSCs for different plant states.

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# BDBEs – Basis

- Inclusion of additional, beyond-design-basis events part of the evolution of Parts 50 and 52. In general, a means to resolve issues with credit given to non-safety-related SSCs.
  - Anticipated transients without scram (ATWS), station blackout (SBO) (Part 50)
  - Severe accident design features (Part 52)
  - Aircraft impact assessments (§ 50.150)
  - Mitigation of beyond design events (§ 50.155)
- IAEA – Design extension conditions
- Historically, BDBEs incorporated into the “design basis” of SSCs (see NEI 97-04, RG 1.186).

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# BDBEs – Basis

- Consideration of event sequences and related special treatment for wide range of event frequencies integral to LMP and preceding methodologies (e.g., Next Generation Nuclear Plant, Modular High Temperature Gas-Cooled Reactors) used to develop Framework A. The event categories support defense-in-depth assessments considering both measures to prevent and mitigate events.
- A benefit of addressing very unlikely event sequences within LMP is that the approach supports a less stylized, bounding-type DBA.
- Rationale for addressing BDBEs will be provided in Statement of Considerations for the proposed rule package.

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# BDBEs – Questions for Stakeholders

- What alternatives might be proposed for addressing very unlikely event sequences under Framework A?

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# Stakeholder Presentations and Discussions



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# Subpart B – ALARA

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# ALARA – Updated Preliminary Proposed Rule Language

§ 53.260 Normal operations.

(a) *Maximum public dose.* Licensees under this part must ensure that normal plant operations do not result in public doses or dose rates in unrestricted areas that exceed the limits provided in Subpart D to 10 CFR part 20.

(b) *As low as reasonably achievable.* A combination of design features and programmatic controls must be established such that the estimated total effective dose equivalent to individual members of the public from effluents resulting from normal plant operation are as low as is reasonably achievable in accordance with 10 CFR part 20.

(similar text for occupational exposures)

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# ALARA – Basis

- Consistent with current requirements in § 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents— nuclear power reactors.” Additional ALARA requirements tied to the initial design of a facility include Appendix I to Part 50; 10 CFR 20.1101, and 40 CFR Part 190 (EPA).
- Consistent with previous design certification applications.

10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors.

(e) Each application for a design approval, a design certification, or a manufacturing license under part 52 of this chapter shall include:

(1) A description of the equipment for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section; and ...

(a) ... a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents ... the application shall also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as is reasonably achievable....

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# ALARA – Basis

- Recognizes that plant design plays essential role in controlling releases and protecting plant workers.
- Consistent with past Commission decisions (Part 20 rulemaking, Advanced Reactor Policy Statement).
- Many cost-effective solutions are most effectively identified and addressed at the design stage of a project.
- Staff is proposing more performance-based approach to preparing applications and NRC review of ALARA during design reviews through issuing draft guidance (Advanced reactor content of application project (ARCAP)).
- Rationale for maintaining ALARA requirements—for both licensees and designers— will be provided in Statement of Considerations for the proposed rule package.

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# ALARA – Questions for Stakeholders

- What alternatives from the existing requirements and recent applications are being contemplated?
- Are some issues not being addressed by the guidance being developed under ARCAP?

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# Stakeholder Presentations and Discussions

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# Subpart F – Facility Safety Program

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# FSP – Updated Preliminary Proposed Rule Language

## § 53.890 Facility safety program.

Each licensee must establish and implement a facility safety program (FSP) that routinely and systematically evaluates potential hazards; operating experience related to plant SSCs, human actions, and programmatic controls affecting the safety functions required by § 53.230; and the resulting changes in risks to the public from operation of the facility over its operating lifetime. An FSP must include a risk-informed, performance-based process to proactively identify new or revised internal or external hazards to the facility and performance issues related to plant SSCs, human actions, and programmatic controls; assess changes in the risks posed to the public from the licensed commercial nuclear plant; and, when appropriate, must consider measures to mitigate or eliminate the resulting risks using the criteria defined in § 53.895. The FSP must be implemented and supported by a written FSP as required in § 53.900.



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# FSP – Basis

- Introduced, in part, to address possibility of smaller but more numerous plant sites. Ongoing assessment of new information may be best performed by those most familiar with systems, hazards, and associated risks.
- More numerous sites with some expectations of changes to regulatory oversight introduces need for alternatives to current NRC-centric approach. Could enable changes to NRC processes related to operating experience, generic safety issues, and backfit analyses.
- Regulatory models assigning responsibility of assessing risks to licensees and providing flexibility in addressing changing risks taken from regulatory theory (Sparrow), NRC Part 70 (§ 70.62), and other agencies (DOT, DOE).
- Practical to implement as part of overall approach with periodic updating of PRA, evaluation of plant changes (§ 50.59 equivalent), and other risk management activities.
- Rationale for proposing FSP will be provided in Statement of Considerations for the proposed rule package.
- Specific question to solicit comments may be included in the proposed rule.

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# FSP – Questions for Stakeholders

1. How could an FSP be considered within an overall model of licensing and regulating future plants?
2. In assessing preliminary proposed rule language, are there suggestions on the performance criteria for evaluating new information and considering risk-reduction measures?

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# Stakeholder Presentations and Discussions

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# Other Topics

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# Next Steps—Future Public Meetings

- The staff will continue to announce public meetings to discuss and receive feedback on various regulatory topics and preliminary proposed rule text.
  - Preliminary proposed rule text will be posted on regulations.gov under docket ID NRC-2019-0062 before the public meetings and in ADAMS at ML20289A534.
- The staff will continue to engage with ACRS.
- Stay informed! Subscribe to GovDelivery:  
<https://service.govdelivery.com/accounts/USNRC/subscriber/new>

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# Final Discussion and Questions



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# Closing Remarks

## Rulemaking Contacts

[Robert.Beall@nrc.gov](mailto:Robert.Beall@nrc.gov)

301-415-3874

[Nanette.Valliere@nrc.gov](mailto:Nanette.Valliere@nrc.gov)

301-415-8462

Regulations.gov docket ID: **NRC-2019-0062**

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# Acronyms and Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ALARA	As low as reasonably achievable
AOO	Anticipated operational occurrence
ARCAP	Advanced reactor content of application project
ATWS	Anticipated transient without scram
BDBE	Beyond design basis event
CFR	Code of Federal Regulations
DBA	Design basis accident
DBE	Design basis event
DEC	Design extension condition
DOE	U.S. Department of Energy

DOT	U.S. Department of Transportation
EAB	Exclusion area boundary
EPA	U.S. Environmental Protection Agency
F-C	Frequency-consequence
FSP	Facility safety program
IAEA	International Atomic Energy Agency
LMP	Licensing Modernization Project
ML	Manufacturing license
NEIMA	Nuclear Energy Innovation and Modernization Act
NO	Normal operations
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation



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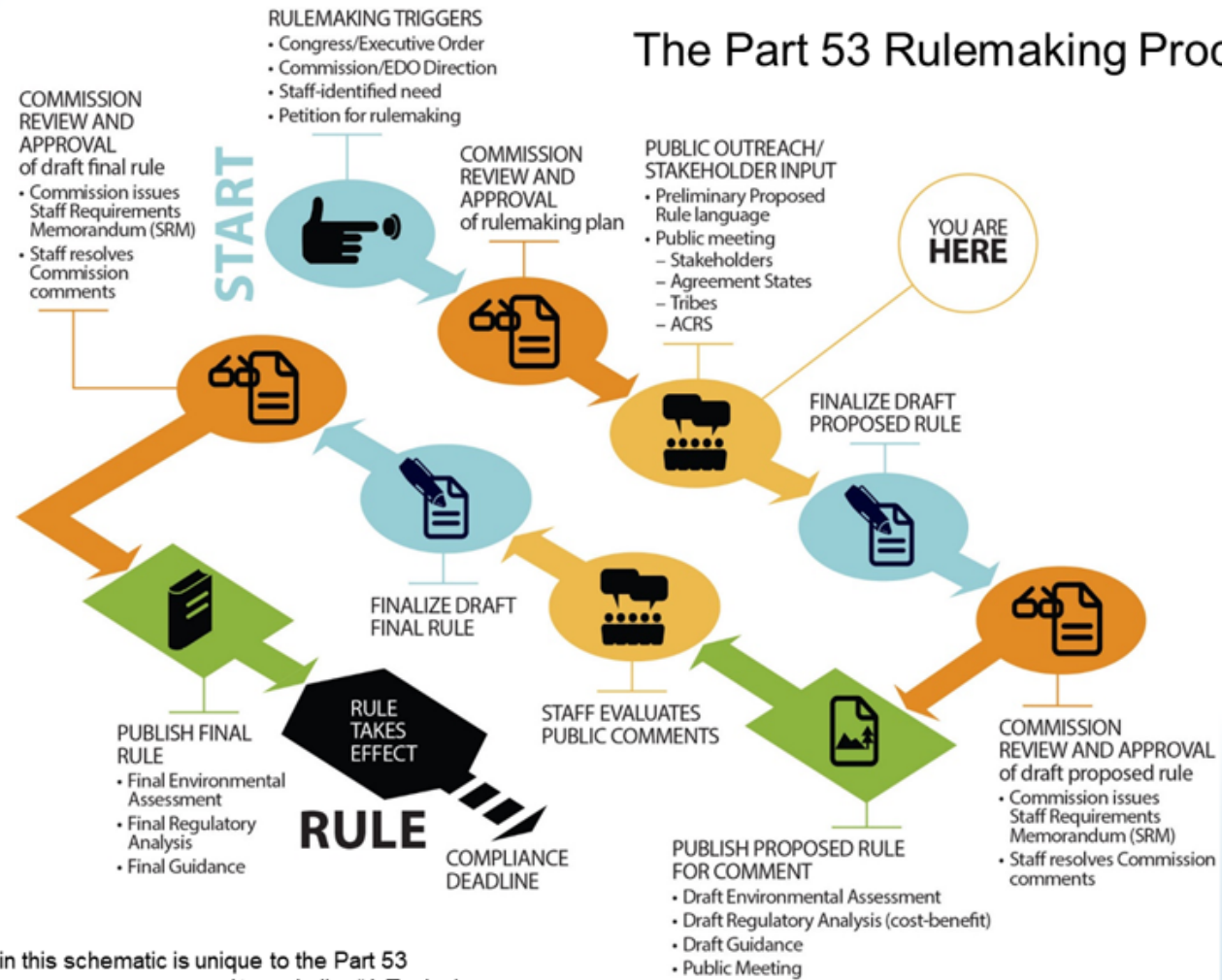
# Acronyms and Abbreviations

NSRSS	Non-safety-related but safety significant
NUREG	U.S. Nuclear Regulatory Commission technical report designation
PAG	Protective action guide
PM	Project manager
PRA	Probabilistic risk assessment
QHOs	Quantitative health objectives
REM	Roentgen equivalent man
RG	Regulatory guide
SBO	Station blackout
SSCs	Structures, systems, and components

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# Background slides

# The Part 53 Rulemaking Process\*



\* The process depicted in this schematic is unique to the Part 53 rulemaking and varies in some ways compared to a similar "A Typical Rulemaking Process" schematic available on the NRC's public website.

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# Rulemaking Schedule

