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

Advanced Reactor Stakeholder Public Meeting

October 12, 2022

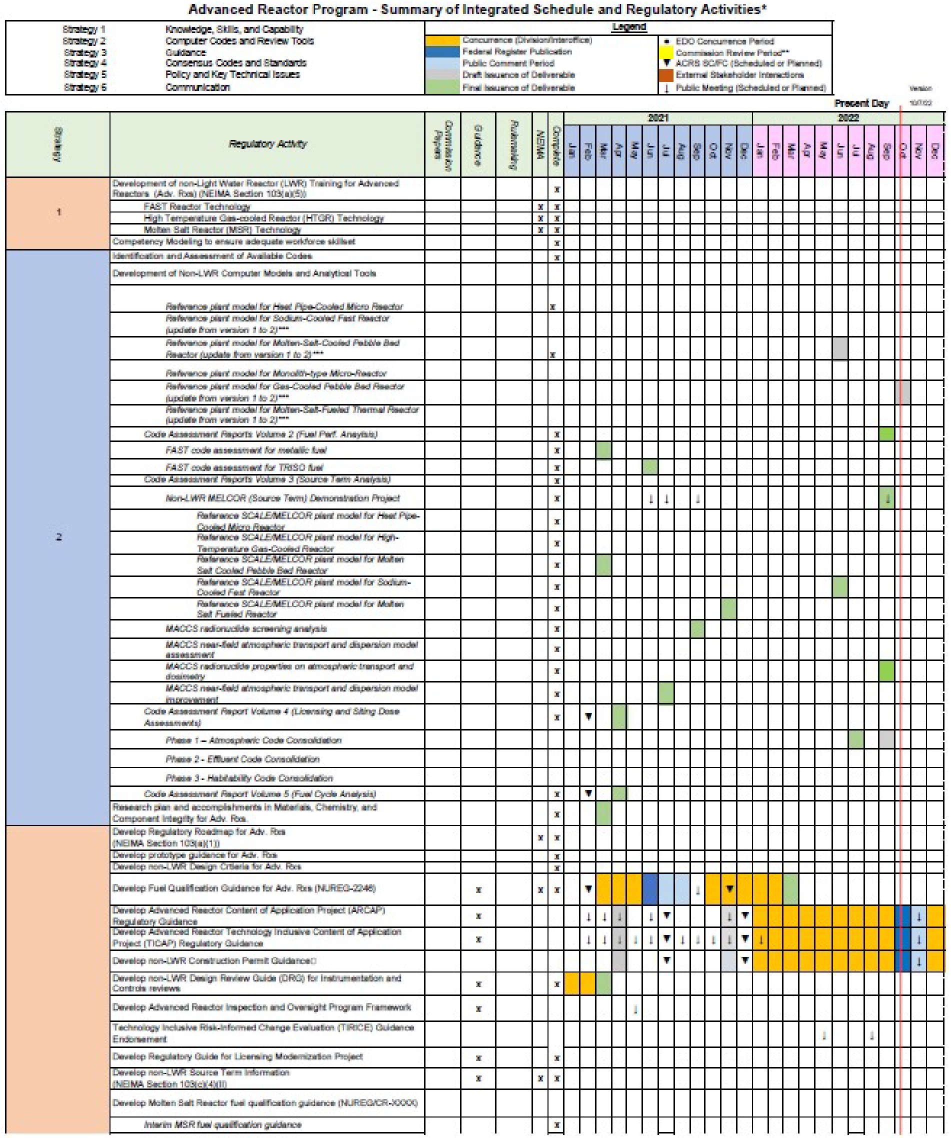
Microsoft Teams Meeting

Bridgeline: (301) 576-2978

Conference ID: 886950993#

Advanced Reactor Program

Advanced Reactor Stakeholder Meeting Agenda October 12, 2022		
Time	**Agenda Topic**	**Presenter(s)**
10:00 – 10:10 am	Opening Remarks Adv. Rx Integrated Schedule	Ossy Font Steve Lynch
10:10 – 10:25 am	CNCS/NRC Third Interim Report	Jeffrey Schmidt
10:25 am – 11:20 pm	Discussion of Draft Outline for Natrium Construction Permit Application	Joseph Sebrosky
11:20 am – 12:00 pm	Part 53 Update: Status and Overview of Revisions; Fire Protection Requirements in Framework B	Bill Jessup Bill Reckley Marty Stutzke Chuck Moulton
12:00 – 1:00 pm	Lunch Break	
1:00 – 2:45 pm	Overview of the Part 53 Subpart F Interim Staff Guidance	Maurin Scheetz, Theresa Buchanan, and Dr. David Desaulniers
2:45 – 2:55 pm	Break	
2:55 – 3:55 pm	Technology-Inclusive, Risk-Informed, and Performance-Based Methodology for Seismic Design of Commercial Nuclear Plants	Jim Xu
3:55 – 4:55 pm	Seismically Isolated Nuclear Plants Guidance	Jim Xu
4:55 – 5:00 pm	Future Meeting Planning and Concluding Remarks	Ossy Font Steven Lynch



https://www.nrc.gov/reactors/new-reactors/advanced/integrated-review-schedule.html

Advanced Reactor Program

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CNSC/NRC TRISO Qualification Assessment

Third Interim Report
Advanced Reactor Stakeholders Meeting

Kelly Conlon, Canadian Nuclear Safety Commission (CNSC)
Jeff Schmidt, U.S. Nuclear Regulatory Commission (NRC)

Memorandum of Cooperation (MOC)

- Generic Tristructural Isotropic (TRISO) qualification assessment is supportive of NRC/CNSC MOC (ML19275D578), Item 2

Area of Cooperation	TRISO Assessment
Development of shared advanced reactor and SMR [small modular reactor] **technical review approaches that facilitate resolution of common technical questions** to facilitate regulatory reviews that address each Participant's national regulations	Exercise the fuel qualification framework developed in Nuclear Energy Agency (NEA) report, "Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors," (ML22018A099) and NUREG-2246, "Fuel Qualification for Advanced Reactors" (ML22063A131)
Collaboration on pre-application activities to **ensure mutual preparedness to efficiently review** advanced reactor and SMR designs	Several proposed advanced reactor designs use TRISO fuel and reference the testing performed as part of the Advanced Reactor Fuel (AGR) program as documented in topical report EPRI-AR-1(NP)-A
Collaboration on research, training, and in the development of regulatory approaches to **address unique and novel technical considerations** for ensuring the safety of advanced reactors and SMRs	Final report will (1) provide evidentiary basis to support regulatory findings for items that are generically applicable to TRISO, (2) identify items that are design dependent, and (3) highlight areas where additional information and/or testing is needed

Assessment Team and Schedule

- Joint report from CNSC and US NRC
- UK regulator, Office for Nuclear Regulation (ONR) involved as an observer
- Technical support provided by Pacific Northwest National Laboratory (PNNL)
- Work plan:

CNSC/NRC Joint TRISO Fuel Assessment Project

Objective/Scope

CNSC and USNRC staff will work together to establish a common regulatory position on TRISO fuel qualification based on existing knowledge and to identify any potential analytical or testing gaps which would need to be addressed to enable TRISO use in advanced reactor licensing applications.

- Available on NRC advanced reactor website

https://www.nrc.gov/reactors/new-reactors/advanced/international-cooperation/collaboration-with-canada.html

☑ **Task A, Project Planning**

- Timeline: Fourth Quarter 2021
- End Product: Initial project plan finalized with resources in place (PNNL contract awarded)

☐ **Task B, Draft Fuel TRISO Fuel Assessment Report**

- Timeline: Fourth Quarter 2021 through Fourth Quarter 2022
- End Product: Four interim draft reports. The final draft will be a comprehensive draft report addressing the goals within the fuel qualification framework from NEA report, “Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors,” and NUREG-2246.

☐ **Task C, Finalize Report**

- Timeline: Fourth Quarter 2022 to Second Quarter 2023
- End Product: The final report will be a joint NRC/CNSC report providing a generic assessment of TRISO fuel

Third Interim Report Will Cover

- Goal was to define SiC end-state properties which yield AGR like fission product retention
- Focus was on AGR-1, Variant 3 and AGR-2 SiC characteristics
- Report examines grain size, grain boundary characteristics, void size, SiC/PyC interfaces
- Third Interim Report is still being finalized therefore the “conclusions” discussed in this presentation are preliminary

SiC End-State Attributes

- Report recognizes the importance of these attributes in retaining fission products
- Generally desirable attributes have been identified
- Definitive ranges of the examined parameters could not be established with high confidence and hence are not suitable as generic licensing criteria
- Identified as a knowledge gap and recommend additional research be performed to characterize these attributes

Preliminary Desirable SiC End-State Attributes

- A reasonably uniform grain size across the SiC layer
- Generally smaller grain sizes are thought to be desirable
- Non-columnar grain boundaries
  - Provides AGR-1 data as examples of grain boundary types
- Proposes a desirable upper limit in SiC void sizes
- Identifies that delamination between the IPyC and SiC is not desirable and recommends additional research/study regarding the interface thickness and morphology

Preliminary Conclusions

- Defining an acceptable range of SiC layer end-state attributes which ensure good (AGR like) fission product retention is desirable but not practical based on currently available information
- Recommends additional research/study to identify acceptable ranges for generic licensing
- The working group seeks stakeholder input which would better define information on relevant SiC parameters and acceptable ranges
- Report provides AGR-1, Variant 3 property values, but doesn't state these are neither sufficient to determine or are necessary to ensure acceptable performance
- The range of acceptable SiC coating parameters is dependent on the applicant licensing needs (i.e., assumed TRISO releases and release pathways) which is usually related to the proposed plant siting

Questions?

(Questions for CNSC should be directed to **mediarelations-relationsmedias@cnsc-ccsn.gc.ca** or by phone at 613-996-6860)

Discussion of Draft Outline for Natrium Construction Permit Application

Natrium Draft Construction Permit Application Table of Contents

- Purpose: To discuss non-proprietary draft Natrium construction permit (CP) application table of contents (TOC) and note differences between Natrium draft CP application TOC and draft advanced reactor content of application project (ARCAP) and technology inclusive content of application project (TICAP) guidance documents
- Outcome: Clear understanding of differences and discussion of whether in the distant future the staff should consider a revision to the TICAP guidance document
- Key documents:
  - Natrium Draft Construction Permit and Preliminary Safety Analysis Report Table of Contents, August 29, 2022 (ML22258A301)
  - Nuclear Energy Institute (NEI) 21-07, Revision 1, "Technology Inclusive Guidance for Non-Light Water Reactors; Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology" (ML22060A190)

ARCAP and TICAP - Nexus

Outline Safety Analysis Report (SAR) – Based on TICAP Guidance

1. General Plant Information, Site Description, and Overview of the Safety Case
2. Methodologies and Analyses and Site Evaluations*
3. Licensing Basis Events
4. Integrated Evaluations
5. Safety Functions, Design Criteria, and SSC Safety Classification
6. Safety-Related SSC Criteria and Capabilities
7. Non-safety related with special treatment 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

Additional Portions of Application

- Technical Specifications
- Technical Requirements Manual
- Quality Assurance Plan (design)
- Fire Protection Program (design)
- Quality Assurance Plan (construction and operations)
- Emergency Plan
- Physical Security Plan
- 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
- Facility Safety Program (under consideration for Part 53 applications)

- Safety Analysis Report (SAR) structure based on clean sheet approach

* TICAP chapter 2 supplemented by ARCAP ISG Chapter 2, "Site Information."
Additional contents of application outside of SAR are still under discussion. The above list is draft and for illustration purposes only.

Natrium Draft Construction Permit Application Table of Contents

General Note

- The NRC staff's observations are limited to the structure of the application
  - Today's discussion is limited to non-proprietary high-level information
    - Outside the scope of this presentation, the NRC staff notes that preapplication activities related to the Natrium review continue
      - Information on Natrium preapplication activities can be found at: https://www.nrc.gov/reactors/new-reactors/advanced/licensing-activities/pre-application-activities/natrium.html

Key Observations

- The Natrium draft CP TOC generally aligns with the draft ARCAP and TICAP guidance
  - Noted differences include:
    - ARCAP Draft White Paper interim staff guidance (ISG), Chapter 2, "Site Information," is included in Section 1.2 of the Natrium TOC
      - The NRC staff notes that in the forthcoming draft ARCAP ISG Chapter 2 update there will be a new subsection on Volcanic Hazards based on RG 4.26 (ML20272A168)

Natrium Draft Construction Permit Application Table of Contents

Key Observations (continued)

- Noted differences (continued):
  - Natrium draft TOC Chapters 6 and 7 differ from NEI 21-07 Revision 1 outline
    - Natrium draft TOC Section 6.4, “Reliability and Capability Targets for NSRST [non-safety-related with special treatment] SSCs [structures, systems and components],” and Section 6.5, “Special Treatment Requirements for NSRST SSCs,” would normally be found in NEI 21-07, Revision 1 Chapter 7
  - Natrium draft TOC Chapter 7, “Descriptions for Safety Significant SSCs,” contains a listing of both SR and NSRST SSCs (the details of what SSCs are SR and what SSCs are NSRST are considered proprietary at this point)
    - Proposed grouping is thought to provide a better integrated discussion of various SSCs and their subsystems
    - The NRC staff believes this approach has merit
      - Detailed discussions on various sections/subsections ongoing (some of this information is proprietary)

Natrium Draft Construction Permit Application Table of Contents

Key Observations (continued)

- Other issues
  - The NRC staff notes that the Natrium draft TOC does not include an item for the fitness for duty construction program requirements (see Subpart K of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26)
    - The NRC staff expects that a forthcoming draft ARCAP Roadmap ISG will include guidance in this area
  - A better understanding of what will be included in SAR Chapter 8, “Plant Programs,” at the CP stage would be helpful
    - NEI 21-07, Revision 1 provides guidance in this area
    - The NRC staff expects that a forthcoming draft ARCAP Roadmap ISG and TICAP DG will include additional CP guidance for SAR Chapter 8

Next Steps

- Detailed discussion of Natrium draft TOC continuing with TerraPower as part of preapplication phase
- Based on differences with treatment of site information (ARCAP ISG Chapter 2), Chapters 6 and 7, and expectations for information in SAR Chapter 8, the NRC staff may consider future revisions to ARCAP/TICAP guidance

Natrium Draft Construction Permit Application Table of Contents

Questions?

Periodic Advanced Reactor Stakeholder Meeting

Part 53 Update

October 12, 2022

Agenda

- Part 53 Rulemaking Schedule
- Licensing Frameworks Overview
- Preliminary Proposed Rule Package
- Recent Changes to Preliminary Proposed Rule Language
- Fire Protection Requirements
- Consideration of Recent Stakeholder Feedback
- Next Steps
- Open Forum

Part 53 Rulemaking Schedule

Part 53 Licensing Frameworks

Subpart A - General Provisions

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

Framework A

- PRA-led approach
- Functional design criteria

Subpart N - Siting
Subpart O - Construction/Manufacturing
Subpart P - Operations
Subpart Q - Decommissioning
Subpart R - Application Requirements
Subpart S - License Maintenance
Subpart T - Reporting
Subpart U - Quality Assurance

Framework B

- Traditional use of risk insights
- Principal design criteria
- Includes an Alternative Evaluation for Risk Insights (AERI) approach

Part 53 Rule Package Overview

- Draft *Federal Register* notice issued on September 30, 2022, to support upcoming ACRS meeting
- Package includes the following draft documents:
  - Enclosure 1A – Preamble discussion (statements of consideration)
  - Enclosure 1B – Section-by-section analysis
  - Enclosure 1C – Parts 2 through Part 53 Framework A
  - Enclosure 1D – Part 53 Framework B through Part 171
  - DG 1413 – Technology-Inclusive Identification Of Licensing Events
  - DG 1414 – AERI Methodology
  - DRO-ISG-2023-01 – Operator Licensing Programs
  - DRO-ISG-2023-02 – Exemptions from Licensed Operator Staffing Requirements Specified in 10 CFR 50
  - DRO-ISG-2023-03 – Development of Scalable Human Factors Engineering Review Plan

Notable Changes to Preliminary Proposed Rule Language

- Addition of §§ 53.000 and 53.010 that outline purpose of Part 53 and establish independence of the two frameworks
- Framework alignment, including common discussions of equivalent subparts in preamble
- Addressed several areas of stakeholder interest
  - Developed strategy for Generally Licensed Reactor Operators (GLROs) in Framework A and extended it to Framework B
  - Streamlined fire protection requirements in Framework B
  - Added new, risk-informed siting requirements in Framework B
  - Developed risk-informed, performance-based seismic design alternatives for Framework B

Fire Protection Requirements

- Rule language
  - Framework B
    - Section 53.4350 is now aligned with Framework A (§ 53.875)
    - Self-contained rather than pointing to other sub-sections since Framework B does not have the same structure as Framework A
  - Framework A (§ 53.450(g)) revised to specifically mention ability to address fires within licensing basis events
- Guidance
  - The staff expects that Regulatory Guides 1.189 and 1.205 as written will serve as the basis for fire protection guidance for Part 53
  - There will be an opportunity for stakeholder engagement in the development of additions and modifications needed for the new rule

Fire Protection Requirements in Framework B

- New structure of § 53.4350
  - (a) Fire Protection Plan
    - High level description of the fire protection requirements
  - (b) Fire Protection Program
    - Requirements for the implementation of fire protection policy
    - Implements FP defense-in-depth
  - (c) Fire Protection Program Performance Criteria
    - Fundamental fire protection design criteria
    - Similar to GDC 3
  - (d) Fire Hazards Analysis
    - Describes the requirements for evaluating the capability of a plant to perform safe-shutdown functions and minimize radioactive releases in the event of a fire

Recent Stakeholder Feedback

Feedback	NRC Staff Perspectives
Objectives for chemical hazard requirements are unclear	Preamble discussion includes amplifying information to address this feedback. Chemical hazards in question would include substances commingled with licensed material or those produced by a reaction with licensed material, consistent with similar requirements in Part 70
Rule language is not technology-inclusive in some areas (e.g., references to MBDBE requirements in § 50.155)	Staff revised several sections to ensure that the proposed rule is technology-inclusive, including MBDBE requirements
PRA development at CP stage is not reasonable	The requirement to have a PRA developed to support a CP application is consistent with the 50/52 rulemaking and other Commission policies
Proposed entry conditions for AERI are too conservative	AERI entry conditions distinguish between plants with relatively straightforward designs and plants with relatively complicated designs that warrant the development of a PRA in order to understand their risk. The proposed AERI option is a departure from current Commission policy, which requires all new plants to have a PRA
Several of the requirements in § 53.4730(a)(12) are not technology-inclusive	These requirements were derived from 50.34(f) and, consistent with the Part 50 requirements, only need to be met if they are "technically relevant" to an applicant's design

Next Steps

- October 18 – 19, 2022: ACRS Subcommittee meeting on Regulatory Rulemaking, Policies, and Practices: Part 53
- November 1 – 4, 2022: ACRS meeting
- February 2023 delivery of proposed rule to Commission
- Summer 2023 issuance of proposed rule package followed by formal public comment period
- Additional public meetings, as necessary, to discuss development of the proposed rule package

Open Discussion

Additional Information

- Additional information on the
10 CFR Part 53 rulemaking is available at
https://www.nrc.gov/reactors/new-reactors/advanced/rulemaking-and-guidance/part-53.html

- For information on how to submit comments go to https://www.regulations.gov and search for Docket ID NRC-2019-0062

- For further information, contact Robert Beall, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-3874; email: Robert.Beall@nrc.gov

LUNCH

BREAK

DRO-ISG-2023-01
Operator Licensing Programs
Draft Interim Staff Guidance

Theresa Buchanan
NRR/DRO/IOLB
October 12, 2022

Purpose

- To assist staff reviews of applications under 10 CFR Part 53 related to the operator licensing examination program.
- To provide guidance for review of tailored initial and requalification examination programs
  - For specifically licensed operators (SROs and ROs)
  - For generally licensed operators (GLROs)
- To address proficiency for SROs and ROs
- To assist staff reviews of exemptions from 10 CFR Part 55 for non-large light water power reactor examination programs

Background

- 10 CFR Part 53 is still under development
  - Guidance in this ISG is subject to change based on rulemaking
- Key documents for Part 53 rulemaking can be found at Regulations.gov under Docket ID NRC-2019-0062

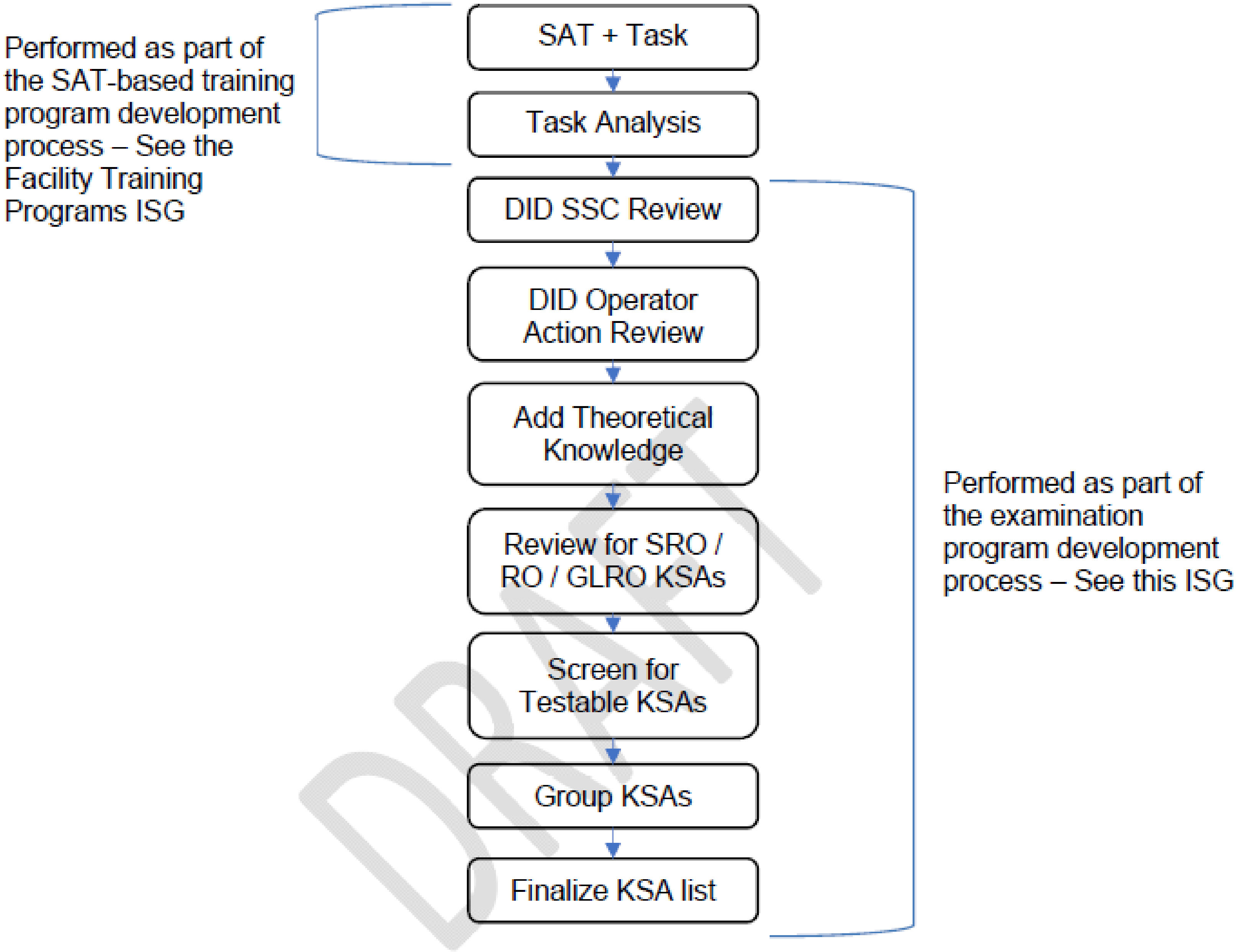
Goals

- Enable facility applicants/licensees to identify knowledge, skills, and abilities (KSAs) necessary for safe operation as the basis for the examination standards
- Establish reliable guidelines for exam program developments based on current best practices from research and expertise on the measurement and testing of KSAs

Section 1.0
KSAs List Development

- Used SAT process to identify a training KSA list
  - This list is not solely limited to tasks related to safe plant operation
  - See DRO-ISG-2023-04, “Facility Training Programs,” and NUREG-0711 for more information
- Using this list as a starting point, screened the list to identify those tasks important to safe plant operation and/or related to the foundational theory of plant operations to develop the KSA list for the exam program
  - Depending on the original list, may have needed to add or remove items to get the necessary KSAs for testing

Figure 1.0 – Overview of KSA Development Process Steps



Section 2.0
Operator Licensing Test Development

- Developed Test Plan
  – How the testable KSAs will be measured
  – For example, what KSAs will be tested using a written test, or a walkthrough format, etc.
  – What the format for the test will be
- Developed detailed content specification
  – What specific KSAs the exam type (written, oral, scenario, JPM, etc) covers
  – How the KSAs are sampled for each examination developed
  – How the test items are reviewed for clarity, quality, and other psychometric issues

Section 3.0
Examination Validity

- Describe validation plan
  - What evidence was collected to support validity of the test, that the test works and will work as intended
    - Content validity, concurrent validity
    - Should require content validity at the least

Section 4.0
Scoring Specifications

- Criterion-referenced
  - Described how each test item is scored and how scores combined to get total score
  - If based on scorer observation, described steps to eliminate any bias in judgments
  - Provided cut-off score

Section 5.0
Reliability of the Test

- If individual repeats the test, the result would be similar to the original result
- Documentation that the tests will have stability of test performance over time
- Documentation of findings that are adequate to justify use of the test for operator licensing

Section 6.0
Test Manual

- Companion to the test plan
- Provides more detail related to the specific types of tests
- Includes administrative aspects of test
  - How to administer
  - Time to administer or time allowed to take the test
  - Materials provided to test takers
  - How to interpret test results

Section 7.0
Additional Characteristics of High-Quality Test Materials

- This section is specifically for written and computer-based tests.
- Provides additional characteristics associated with psychometrics, test instructions, objective scoring system, and standardization

Section 8.0
Other Examination Program Considerations

- This section references back to sections of NUREG-1021, “Operator Licensing Examination Standards for Power Reactors” for items that are universally applicable, regardless of plant design

Section 9.0
Simulation Facilities

- Documentation on how the simulation facility provides a level of fidelity sufficient to assess KSAs as required by 10 CFR Part 53.780(e) or 53.815(e)
- Simulation facilities should have same cognitive requirements as the real environment.
- For simulation-based assessment, documentation provided on how that examination is valid

Section 10.0
Administering Operating Tests

- Examination procedures should be similar to those in NUREG-1021, as specific to the type of test administered
- Measures are in place to ensure examiners behave in accordance with codes of conduct to ensure examination integrity
- Measures are in place to retain required records

Section 11.0
Examination Program Change Management Process

- Documentation specifies what changes require NRC approval and which do not
  - NRC approval
    - Exemption from regulation
    - Change to technical specification
    - Negative impact to examination security/integrity
    - Negative impact on consistency

Section 12.0
Static Computer-Based Testing

- Beyond the scope of the guidance
- The documentation would need to describe how this approach is equivalent to the guidance provided in the ISG

Section 13.0
Additional Guidance for Requalification Programs

- Any requalification failures must be remediated and retested prior to returning to license duties
- For ROs and SROs
  - Periodicity not to exceed 24 months
- For GLROs
  - Periodicity defined by program
  - If >24 months, bases provided

Section 14.0

Proficiency Programs for Specifically Licensed Operators and Senior Operators

- Actively perform the functions
- Maintain proficiency and familiarity
- Re-establish proficiency if it cannot be maintained

Section 15.0
Waivers for GLROs

- Appropriate criteria to waive requirements for an examination included in the program
- If similar to 10 CFR 55.47, no further NRC review
- Else, a basis is provided that describes how the criteria ensures individuals are able to safely and competently operate the facility

Appendix A
Currently Approved Examination Methods

- Methods currently approved in NUREG-1021 can be used without needing further basis from the facility or additional NRC review
- Example: use of a 4-part multiple choice written examination with 80% cut score

DRO-ISG-2023-02

Interim Staff Guidance Augmenting

NUREG-1791, 'Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m),' for Licensing Plants under Part 53

ADAMS Accession No. ML22266A068

Maurin Scheetz

U.S. NRC NRR/DRO

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Overview

- Background
  - 10 CFR 50.54m
  - NUREG-1791
  - Experience with review of Small Modular Reactor staffing plans
- Part 53 approach to staffing
- Overview of draft DRO-ISG-2023-02

Background: Current Practice

- Current 10 CFR 50/52 staffing requirement (i.e., 50.54(m)) is prescriptive
- NRC reviews exemptions to this requirement using NUREG-1791, Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)
  - Developed with advanced reactors in mind
  - Performance-based process for determining appropriate number of licensed control room operators
  - 11 steps including a staffing plan validation
- Staff used NUREG-1791 to evaluate novel control room staffing models for NuScale SMR design and concept of operations
- Cannot use NUREG-1791 as written for Part 53 staffing plan reviews because it relies on exemptions to Part 50 requirements

Part 53 Approach to Staffing

- Applicant proposes minimum staffing level by submitting a staffing plan with application
- Consider differences in staffing level when operators have/do not have a safety role (i.e., for specific or generally licensed operators) – if specific licenses then applicants must include more detail supported by HFE analysis and assessments
- Operators may fill multiple roles (e.g., maintenance, radiation protection, etc.) so must include these responsibilities in staffing plan submittal
- The staff will review and approve the staffing plan. Changes to approved staffing plans are subject to administrative controls.

Proposed Part 53 Staffing Requirement

Staffing plan. A staffing plan must be developed to include the numbers, positions, and qualifications of operators and senior operators or, if applicable, generally licensed reactor operators across all modes of plant operations, and the numbers, positions, and responsibilities of personnel providing support in areas such as plant operations, equipment surveillance and maintenance, radiological protection, chemistry control, fire brigades, engineering, security, and emergency response.

proposed § 53.730(f)

Proposed Part 53 Requirement for On-Shift Engineering Expertise [§ 53.730(f)(1)]

- The staffing plan must include a description of how engineering expertise will be available to the on-shift crew during all plant conditions to assist in situations not covered by procedures or training
- A person available to support the crew at all times. This person is familiar with the operation of the facility and has a technical degree:
  - bachelors in in engineering or,
  - Bachelors in engineering technology or a physical science or,
  - PE license
- Basis: Commission policy for, "Education for Senior Reactor Operators and Shift Supervisors at Nuclear Power Plants," (published in the Federal Register (54 FR 33639) on August 15, 1989)

DRO-ISG-2023-02: for review of Part 53 staffing plans

- Objective is to guide reviewer through the process of:
  - Evaluating staffing plans and support analyses submitted under § 53.730(f)
  - Determining whether the proposed minimum staffing level provides assurance that plant safety functions can be maintained across all modes of plant operations
  - Approving staffing plans
- For plants that will have specifically licensed operators; could scale the review for plants with generally licensed operators
- Use in conjunction with NUREG-1791
- 11 steps that rely on other Human Factors elements
- Includes review guidance for engineering expertise requirement
- Developed as an Interim Staff Guide (ISG)
  - Following experience with using the ISG the staff plans to update NUREG-1791

DRO-ISG-2023-02: for reviewing engineering expertise

- Guidance on what staff will look at for satisfying engineering expertise requirement to include:
  - Education prerequisites
  - Training and qualification
  - Responsibilities of the job
  - Data needs if offsite
  - Response time if on site
  - Expectations for one or multiple people filling the job
  - Communication needs
  - Cybersecurity expectations
  - Include job in validation activities

DRO-ISG-2023-03

Development of Scalable Human Factors Engineering Review Plans

David R Desaulniers, PhD
Senior Technical Advisor for Human Factors and Human Performance Evaluation
Office of Nuclear Reactor Regulation / Division of Reactor Oversight
USNRC
October 12, 2022

Overview

- Background
  - NRC's human factors engineering (HFE) regulatory basis
  - Review practices
  - Recent experience
- Overview of scalable HFE review process
- Overview of draft DRO-ISG-2023-03

Background: Current Practice

- Current 10 CFR 50 HFE requirement (i.e., 50.34(f)(2)(iii)) is focused on the main control room
- NRC's HFE reviews for large light-water reactors have been conducted using NUREG-0711, Human Factors Engineering Program Review Model
  - Systems engineering based approach
  - 12 program elements and 300+ criteria
- Lessons-learnt from recent Part 52 reviews indicated a need for a new approach to regulation and review of HFE for advanced reactor technologies

Background: Proposed Part 53 Approach to HFE

- HFE to be required where necessary to support important human actions
- HFE reviews to be application specific (i.e., scaled) considering the characteristics of the facility design and its operation

Background: Proposed Part 53 HFE Requirement

The plant design must reflect state-of-the-art human factors principles for safe and reliable performance in all locations that human activities are expected for performing or supporting the continued availability of plant safety or emergency response functions.

[proposed (§ 53.730(a)]

Background: Draft Guidance

- Objective is to guide reviewer through the process of:
  - Developing an application specific review plan
  - Identifying appropriate HFE review guidance
- To be used in place of NUREG-0800, Chapter 18, Human Factors Engineering
- Developed as an Interim Staff Guide (ISG)
  - Following experience with using the ISG the staff plans to make the guidance a NUREG

Scaling Process: Overview

- Begins - during pre-application engagements (if conducted)
- Concludes - with completion of application acceptance review
- Conducted - in 5 steps leading to the staff assembling the review plan

Scaling Process: 5 Steps

1. **Characterization** – establishing a documented understanding of the design and its operation from an HFE perspective
2. **Targeting** – identifying aspects of the design and operation for HFE review
3. **Screening** – selecting HFE program elements / activities for review in conjunction with each target
4. **Grading** – selecting specific standards and guidance documents to be applied to the review
5. **Assembling the review plan** – integrating results of prior steps to produce a plan that supports an efficient, risk-informed, reasonable assurance determination

Scaling Guidance: Overview

- Main body (22 pages) – provides essential guidance for developing the review plan
- Appendices (88 pages) – provide supporting guidance for implementing each step of the process

Scaling Guidance: Main Body – Key Features

- Applicability:
  - Standard Design Approvals (SDAs), Design Certifications (DCs), Combined Licenses (COLs) and Operating Licenses (OLs)
- Rationale for scaling reviews
- Regulatory basis / acceptance criteria
- Guidance for each step of scaling process
  - Objective
  - Process
  - Reviewer Responsibilities
- Focus is on “what to do / accomplish” when scaling reviews

Scaling Guidance: Appendices – Key Features

- Focus is on “how to”
- Recommended methods for each step of scaling process
- Pointers to sources of additional guidance

Scaling Guidance: Appendix A

Characterization:

- What to include in the characterization – essential elements
- How to organize and document the characterization
- Use of the characterization to aid coordination with related reviews (e.g., staffing, operator licensing, I&C)

Scaling Guidance: Appendix B

Targeting:

- General principles for target selection
- Descriptions of 38 prospective (example) characteristics of advanced reactor designs and operations
  - Human performance implications
  - Availability of guidance to support reviews

Scaling Guidance: Appendix C

Screening:

- General strategies and specific considerations for selecting which HFE activities to review or screen out
- Implications / challenges of advanced reactor design characteristics for certain HFE activities or their review

Scaling Guidance: Appendix D

Grading:

- Guidance for selection of standards and guidance documents to support the review
  - Considerations for use of documents that lack prior NRC endorsement
- Reference table of HFE standards and guidance documents in both nuclear and non-nuclear domains

Scaling Guidance: Appendix E

Assembling the Review Plan:

- Strategies for integrating the results of Steps A-D to develop a plan that is efficient yet sufficient to support a reasonable assurance determination
- Guidance for documenting the review plan and gaining management approval

Thank You

Acronyms Used

- CFR – Code of Federal Regulations
- COL – combined license
- DC – design certification
- DRO – Division of Reactor Oversight
- HFE – human factors engineering
- I&C – instrumentation and control
- ISG – interim staff guide
- OL – operating license
- SDA – standard design approval

NLWR Stakeholders Meeting: Considerations for Flexible Seismic Design Options Under Part 53

Dr. Jim Xu, Senior Level Advisor for Seismic and Geotechnical Engineering

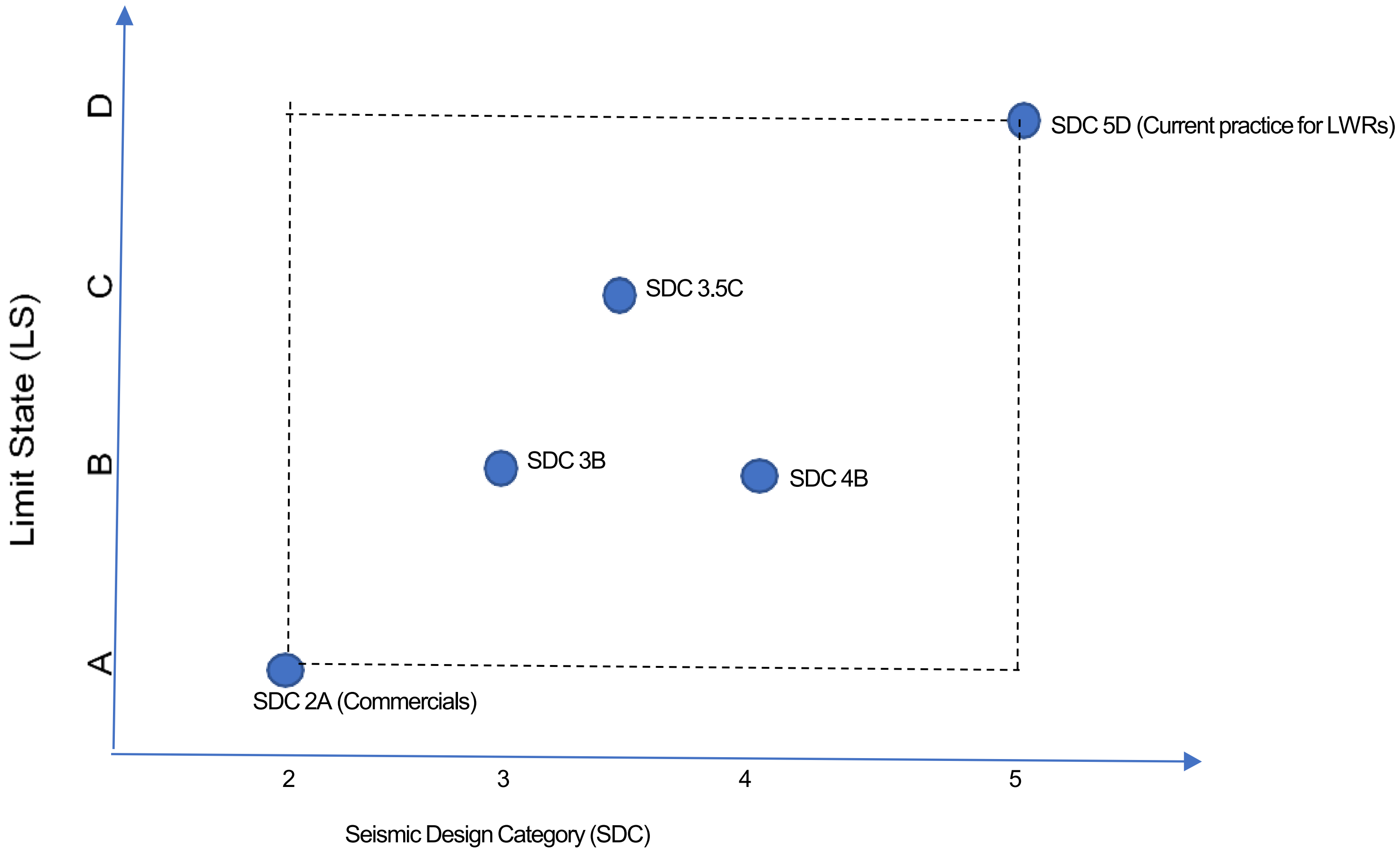
USNRC/RES

October 12, 2022

Topics

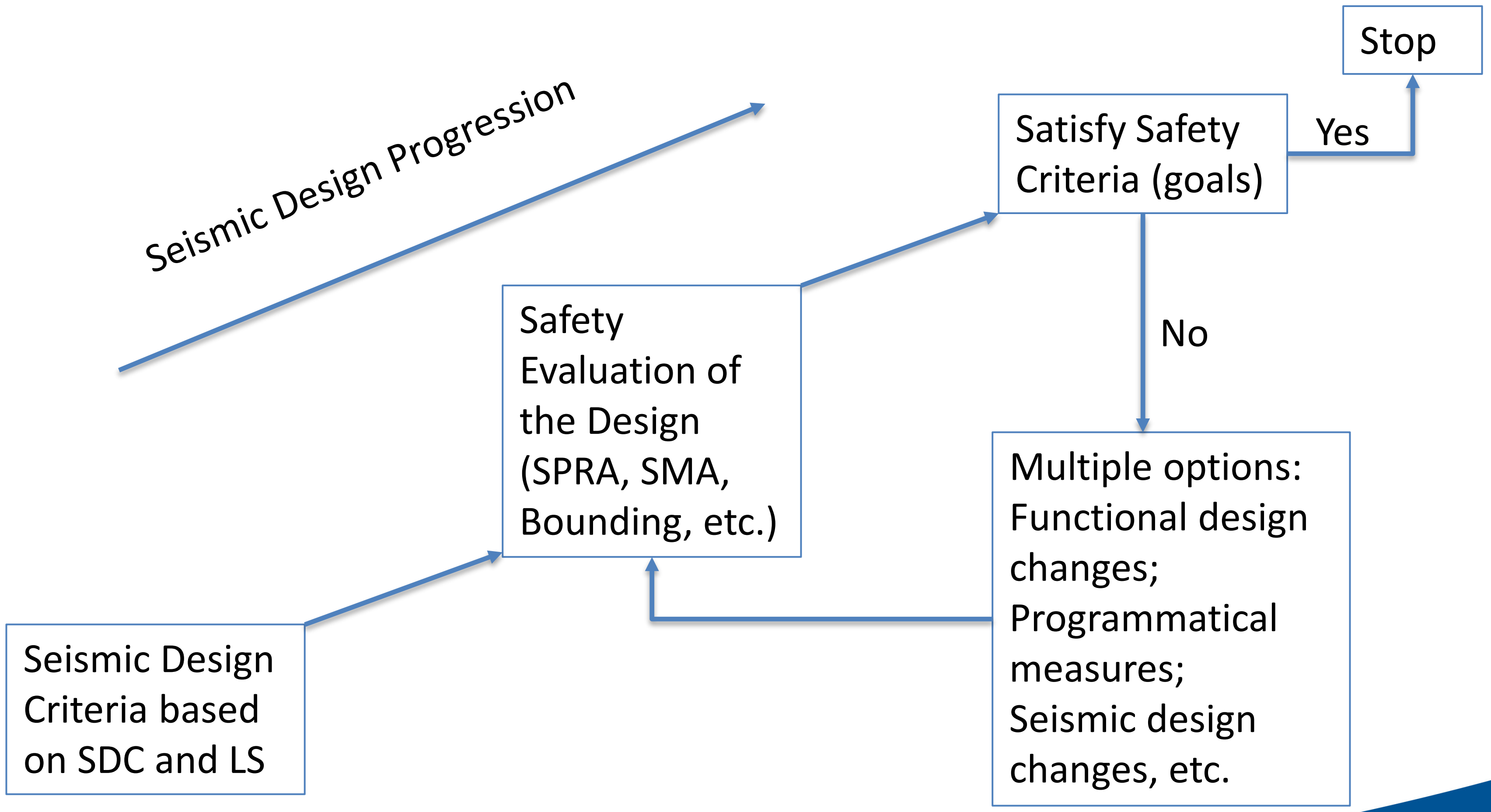
- Pre-decisional draft guide on Technology-Inclusive, Risk-Informed, and Performance-Based Methodology for Seismic Design of Commercial Nuclear Plants (ML22276A149)
- Pre-decisional draft guide on Seismically Isolated Nuclear Plants (ML22276A154)

Fundamental for Seismic Design



A – short of collapse B – large inelastic C - limited inelastic D – essentially elastic

Current Design Practice



Current Design Practice (cont'd)

- Seismic Design Criteria – Step 1
  - Seismic design classifications based on RG 1.29
    - Seismic Category I
    - Nonseismic Category I (potential for II/I interactions)
  - SRP acceptance criteria - §52.47 (a) (9)
- Safety evaluation of the design – Step 2
  - SPRA/PRA-based SMA guidance in ISG-20
- Safety Criteria (goals) – Step 3
  - Surrogates: CDF and LERF
  - Safety margin established using ISG-20 is viewed to meet NRC safety expectations either in terms of surrogates or consequences (QHOs) for LWR seismic design under Part 50/52

Topics

- **Pre-decisional draft guide on Technology-Inclusive, Risk-Informed, and Performance-Based Methodology for Seismic Design of Commercial Nuclear Plants – Framework A of Part 53**
- Pre-decisional draft guide on Seismically Isolated Nuclear Plants

Proposed Framework A of Part 53 Related to Seismic Design

- §53.480 Earthquake engineering
  - Design basis ground motions (DBGMs) (in lieu of single SSE in Part 50)
- §53.450 Analysis Requirements
  - Probabilistic risk assessment (PRA)
  - DBA assessment
- Safety Criteria
  - §53.21 0 (design basis accidents, 25 ram)
  - §53.220 (consequence-based QHOs: 5x10E-7 for prompt fatalities, 2x10E-6 for cancer fatalities)
  - §53.470 (analytical safety margin as alternative but more restrictive than §53.220)

Proposed Pre-decisional DG Options for Seismic Design (Framework A)

- Option 1, based on the current practice
  - Seismic Design Criteria - §53.480
    - Classifications
      - Safety related (SR)
      - Non-safety related but safety significant (NSRSS)
      - Non-safety significant (NSS) (Potential for interactions with SR and NSRSS SSCs)
    - SRP acceptance criteria or ASCE 43-19 criteria corresponding to SDC 5 and LS-D
    - Single DBGM developed based on §53.480
  - Safety evaluation of the design to meet §53.450
    - SPRA/PRA-based SMA guidance in ISG-20 (event sequence development may need to reflect release end states or other risk metrics)

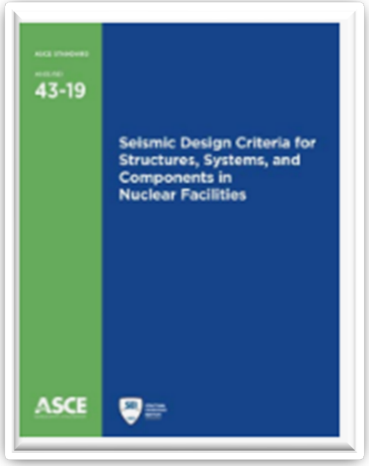
Proposed Pre-decisional DG Options for Seismic Design (Framework A)

- Safety Criteria
  - Safety margin established using ISG-20 is viewed to meet the safety criteria §53.210 and for §53.220 for the seismic design

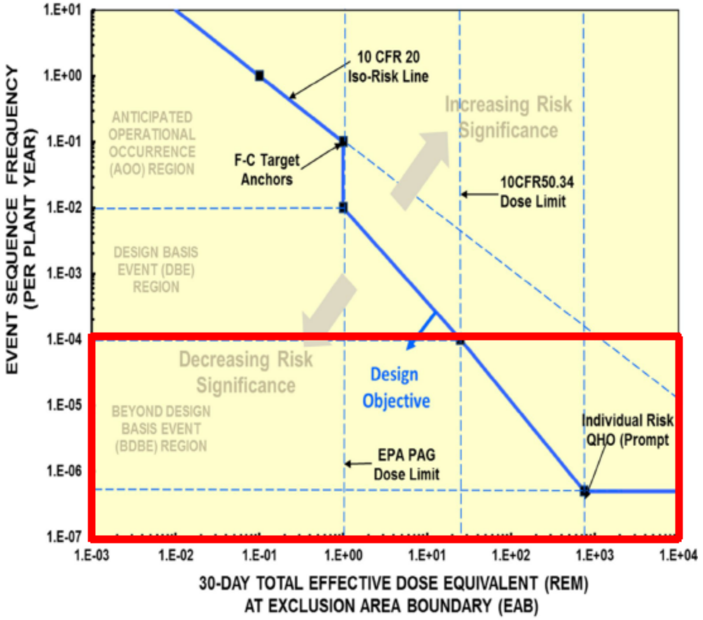
Proposed Pre-decisional DG Options for Seismic Design (Framework A)

- Option 2, based on LMP framework
  - Seismic Design Criteria
    - Seismic design based on ASCE 43-19 performance-based approach
      - SR and NSRSS SSCs can be designed to different SDCs and LSs consistent with their contribution to safety
    - Full utilization of DBGMs per §53.480
  - Safety evaluation of the design to meet §53.450
    - SPRA is fully integrated with the design process
    - DBA assessment
  - Safety Criteria
    - LMP criteria encompasses safety criteria §53.210 and for §53.220

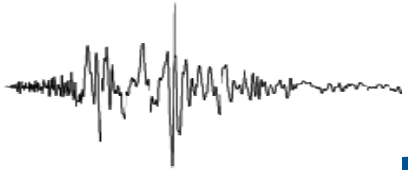
Performance-based Framework



- Design based on performance goal and limit state
- Graded approach to design and analysis
- Risk assessment is more integrated in the design process



- NEI 18-04 and RG 1.233 Licensing modernization project (LMP)
- Frequency-consequence design target
- PRA to quantify risks



Proposed Pre-decisional DG Options for Seismic Design (Framework A)

- Option 3, based on traditional PRA
  - Seismic Design Criteria (similar to Option 2)
    - Seismic design based on ASCE 43-19 performance-based approach
      - SR and NSRSS SSCs can be designed to different SDCs and LSs consistent with their contribution to safety
    - Full utilization of DBGMs per §53.480
  - Safety evaluation of the design to meet §53.450
    - SPRA is fully integrated with the design process
    - DBA assessment
  - Safety Criteria
    - §53.210
    - §53.220 or §53.470

Issues with Implementing Flexible Options for Framework B

- Framework B relies on principal design criteria (PDCs) to ensure safety:
  - Qualitative (e.g., PDC 2 is same as GDC 2 for seismic)
  - Lack of quantitative safety criteria similar to Framework A creates difficulty for implementing the flexible design options
  - If we adopt safety criteria of Framework A, then Options 2 and 3 developed for Framework A also apply to Framework B

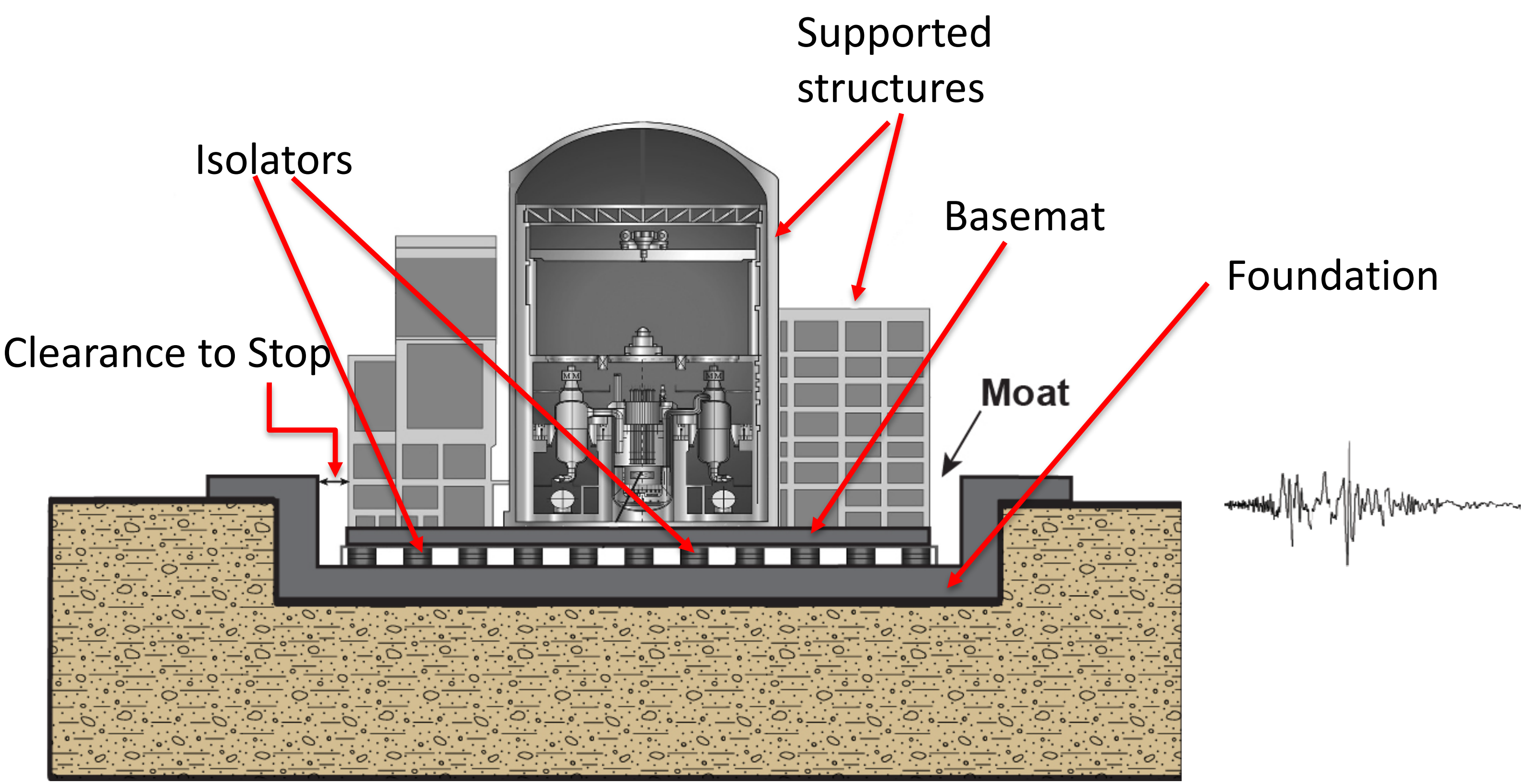
Issues with Implementing Flexible Options for Part 50/52

- Regulatory restrictions on Options 2 and 3:
  - Singular SSE applies to all seismic Category I SSCs
  - Appendix S minimum ground motion of 0.1g at foundation level applies
- If we select SDC and LS that are less than SDC 5 and LS-D, we need safety criteria similar to Framework A

Topics

- Pre-decisional draft guide on Technology-Inclusive, Risk-Informed, and Performance-Based Methodology for Seismic Design of Commercial Nuclear Plants – Framework A of Part 53
- **Pre-decisional draft guide on Seismically Isolated Nuclear Plants – Framework A**

Concept

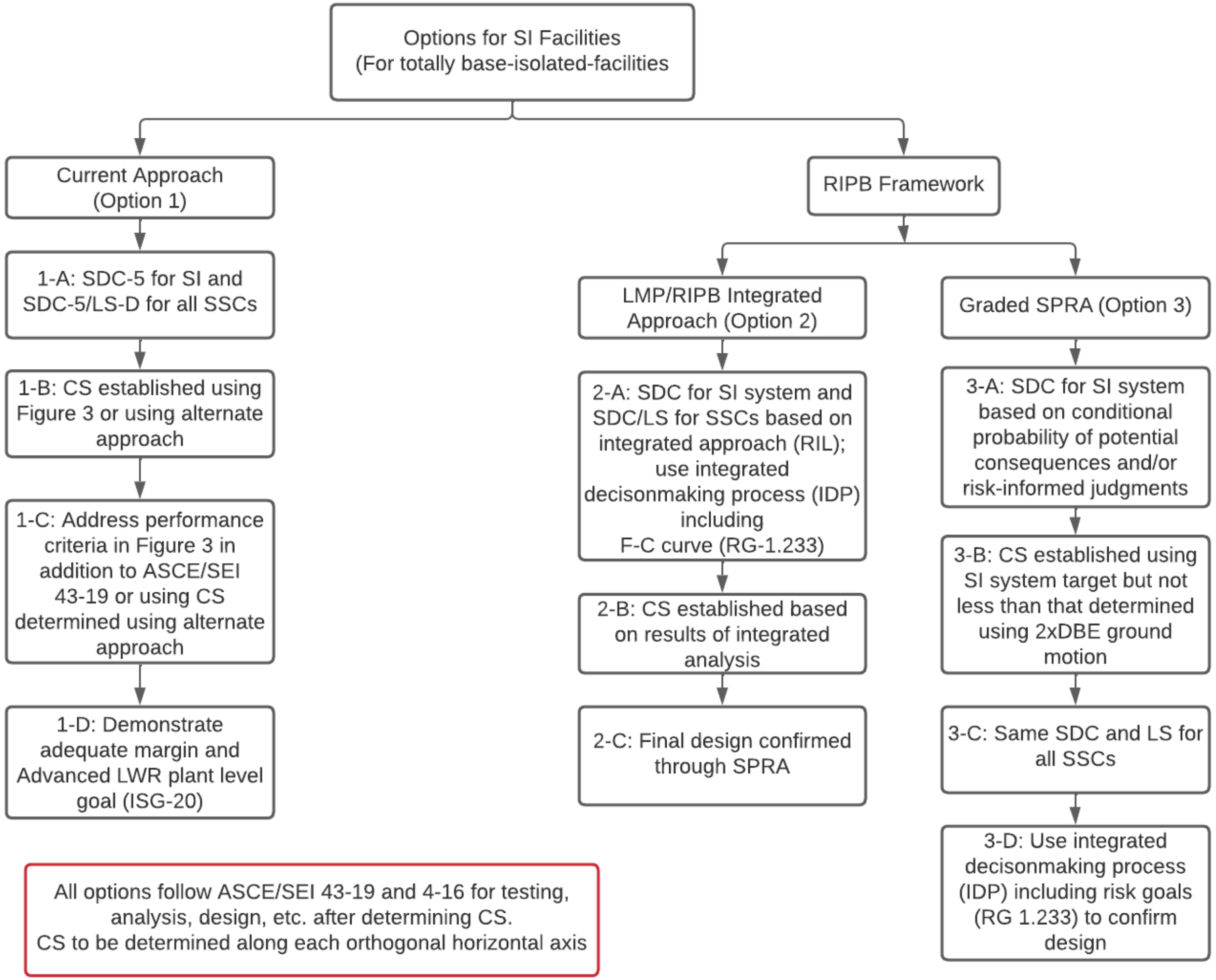


Supported structures
Isolators
Basemat
Foundation
Clearance to Stop
Moat

Pre-decisional DG (Framework A)

- Technical considerations:
  - Use the same technical approach as described in Pre-decisional RIPB DG (3 options)
  - Focus on addressing SI specific criteria for each of 3 options
  - Guidance relies on ASCE 43-19 and ASCE 4-16 as well as available literature

Three Options



Performance Criteria for Option 1

Figure 3: Performance Criteria for Adopting Seismic Isolation Systems for Commercial Nuclear Power Plants[1]					
	Isolation System		Superstructure design and performance	Umbilical line design and performance	Moat or stop design and performance
Ground motion levels	Isolator unit and system design and performance criteria	Approach to demonstrating acceptable performance of an isolator unit			
DBGM corresponding to SDC 5	No long-term change in mechanical properties. 95% confidence of the isolation system surviving without damage when subjected to the mean displacement of the isolator system under the DBGM loading.	Perform production testing on each isolator for the mean system displacement under the BDGM loading and corresponding axial force.	Superstructure design and performance to conform to current seismic design practice in SRP for DBGM loading after filtering through the seismic isolation system.	Umbilical line design and performance to conform to SRP for DBGM loading.	Moat gap sized such that there is less than 1% probability of the superstructure impacting the moat or stop for DBGM loading.
BDBE DBGM represents the envelope of 167% of the DBGM	90% confidence of each isolator and the isolation system surviving without loss of gravity-load capacity at the mean displacement under BDBE DBGM loading.	Perform prototype testing on a sufficient number of isolators at the clearance to the stop (CS) displacement and the corresponding axial force to demonstrate acceptable performance with 90% confidence. Limited isolator unit damage is acceptable but load-carrying capacity must be maintained.	Less than a 10% probability of the superstructure contacting the moat or stop under BDBE BDGM loading.	Greater than 90% confidence that each type of safety-related umbilical line, together with its connections, shall remain functional for the CS displacement. Performance may be demonstrated by testing, analysis or a combination of both.	Moat gap sized such that there is less than a 10% probability of the superstructure impacting the moat or stop for BDB BDGM loading. Stop designed to survive impact forces associated with isolation system displacement to 95th percentile BDBE BDGM isolation system displacement.[2] Limited damage to the moat or stop is acceptable but the moat/stop should perform its function.

1.Criteria developed in this Table used the supporting information documented in NUREG/CR-7253.
2. Impact velocity calculated at the displacement equal to the CS assuming cyclic response of the isolation system for motions associated with the 95th percentile (or greater) BDB DBGM displacement.

Summary

- Proposed 3 flexible options for seismic design under Framework A in RIPB pre-decisional DG
- Discussed challenges for implementing the flexible options under Framework B and Part 50/52
- Proposed to use the same flexible RIPB options to address seismically isolated NPPs under Framework A
- SI pre-decisional DG focused on addressing SI specific safety issues and associated criteria.

Questions?

CLOSING REMARKs