Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: 737th Meeting Advisory Committee On Reactor Safeguards (ACRS)

Docket Number: N/A

Location: Rockville, Maryland

Date: July 8, 2026

Work Order No.: NRC-0707 Pages 1-111

NEAL R. GROSS AND CO., INC.
Court Reporters and Transcribers
1716 14th Street, N.W.
Washington, D.C. 20009
(202) 234-4433

DISCLAIMER

UNITED STATES NUCLEAR REGULATORY COMMISSION'S ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The contents of this transcript of the proceeding of the United States Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards, as reported herein, is a record of the discussions recorded at the meeting.

This transcript has not been reviewed, corrected, and edited, and it may contain inaccuracies.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

+ + + + +

737TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

WEDNESDAY

JULY 8, 2026

+ + + + +

The Advisory Committee met at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, and via videoconference, at 8:30.m. EDT, Gregory H. Halnon, Chair, presiding.

COMMITTEE MEMBERS:

GREGORY H. HALNON, Chair

DAVID A. PETTI, Vice Chair

CRAIG D. HARRINGTON, Member-at-Large

ANNIE M. KAMMERER

WALTER L. KIRCHNER

ROBERT P. MARTIN

SCOTT P. PALMTAG

THOMAS E. ROBERTS

MATTHEW W. SUNSERI

DESIGNATED FEDERAL OFFICIAL:

LARRY BURKHART

WEIDONG WANG

ALSO PRESENT:

KEVIN COYNE, NRC

DANIEL DOYLE, NRC

KEVIN HELLER, NRC

JOHN HUGHEY, NRC

JON FACEMIRE, Nuclear Energy Institute (NEI)

MIKE FRANOVICH, NRC

EDWIN LYMAN, Union of Concerned Scientists

NIKO MCMURRAY, ClearPath

CHARLES MOULTON, NRC

TRACY RADEL, NRC

JACQUELINE TULIP, NRC

DEANNA ZHANG, NRC

P-R-O-C-E-E-D-I-N-G-S

8:30 a.m.

CHAIR HALNON: This is the first day of the 737th meeting of the Advisory Committee on Reactor Safeguards. I am Greg Halnon, Chairman of the Committee. ACRS members in attendance in person are Craig Harrington, Annie Kammerer, Walt Kirchner, Bob Martin, Scott Palmtag, Tom Roberts and Matt Sunseri. Member Dave -- or Vice Chair Dave Petti is attending virtually. If I have missed anyone, either ACRS members or consultants, please speak up now.

Weidong Wang and I guess Larry Burkhart are the Designated Federal Officers for today's full committee meeting. We do have a quorum for today's meeting.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act. Under the Atomic Energy Act, ACRS must advise the Nuclear Regulatory Commission on the hazards of proposed and existing reactor facilities and the adequacy of proposed safety standards. Following issuance of Executive Order 14300, the Committee focuses on those activities necessary to fulfill its statutory obligations.

As a result, ACRS is prioritizing the review and reporting of new reactor facilities and new or revised proposed safety standards with particular attention to issues that are unique, novel, and noteworthy. The Committee will consider other nuclear safety matters at the direction of the Commission. Please note, the ACRS speaks only through its published letter reports. All Member comments should be regarded as only the individual opinion of that Member, not a Committee position.

Information about ACRS activities, such as letters, meeting rules and transcripts are available on the NRC public website. The ACRS provides an opportunity for public comment and input on relevant technical issues during our proceedings. For this Full Committee meeting we have received no written statements or requests to make verbal comments. Written statements may be still forwarded to today's Designated Federal Officer, and I'll go through some additional information on that at the end of my comments. We have set aside time in this meeting for public verbal comments as well.

A transcript of the Committee is being kept and will be posted on our website. When addressing

the Committee, the participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. If you are not speaking, please mute your computer on Teams. If you are participating by phone, press star-6 to mute your phone and star-5 to raise your hand on Teams.

The Teams chat feature is only for communicating IT issues or brief meeting logistics. Please do not use it for comments or questions on the topics under Committee discussion. For everyone in the room, please put all your electronic devices in silent mode and mute your laptop microphone and speakers. In addition, please keep sidebar discussions in the room to a minimum since the ceiling microphones are live.

For the presenters, your table microphones are unidirectional, and you'll need to speak into the front of the microphone to be heard online. And we will let you know if for some reason we can't hear you. Finally, if you have any feedback for the ACRS about today's meeting, please fill out the public meeting feedback form on the NRC's website.

Our first item for this Committee meeting is going to be the Proposed Rulemaking on Modernizing

Reactor Licensing, Safety Oversight, and Siting Practices Related to Executive Order 14300. Before I turn it over to Vice Chair Dave Petti, the Subcommittee Chair for this rule, to lead the discussion, I wanted to highlight a few important points about public comments.

First, today's agenda is general. Most of our time will be spent in Committee deliberation on a rule. We will begin with a short presentation from the NRC staff followed by an opportunity for public comments to the Committee. The public comment portion will be limited to one-and-a-half hours so the Committee can proceed with letter writing.

Individual commenters will be asked to limit remarks to three minutes or less. Second, I want to clarify that any public comments that are made are solely for the benefit of the ACRS deliberations. They will not be considered part of the NRC staff's rulemaking record.

If you want your comments considered in the rulemaking, please submit them electronically using the regulations.gov website. And I believe this rule will be posted on that soon. Third, if you wish to make a public comment to the Committee, if you have

time and are able, please let us know by emailing Larry Burkhart at Lawrence.Burkhart, B-U-R-K-H-A-R-T, @nrc.gov and Weidong Wang at Weidong.Wang@nrc.gov, W-E-I-D-O-N-G, dot, W-A-N-G, @nrc.gov, so we can create a list of speakers. Larry and Weidong's email address are also posted in the Teams chat.

After we've gone through the speaker list, we will take additional comments if time permits using the raised hand feature in Teams. Finally, in the coming weeks, the NRC staff will hold a public meeting regarding this proposed rule, and those details will be on the NRC website.

So this will not be the only time you have an opportunity to make comments about the rule. First, before I turn it over to Dave, anything I missed or any comments from the members? Larry, did we get all the information we need? Good.

All right. Just for your information, I did turn the cameras on in the room mainly because we have so many remote participants. So keep that in mind as you're speaking. Dave, I'm going to turn the meeting over to you.

VICE CHAIR PETTI: Thanks, Greg. So we're going to hear a lot about a lot of changes to

the rules. The breadth and depth of the scope that we'll be hearing about is quite a bit. I want to thank the staff before they start. This has been a herculean effort and I think people will get a sense of that when we get into it. So, with that, NRC management want to say something before we begin?

MR. FRANOVICH: Sure. Good morning, Chairman and members of the ACRS. I'm Mike Franovich, incoming Director of the NRC's Office of Nuclear Regulator Research and one of the senior executive sponsors for this rulemaking. Thank you for the opportunity to brief the full committee today on the proposed rule of modernized reactor licensing safety oversight and siting practices.

We call this rule in particular amongst the staff in shorthand as the reactor power to the grid rule. This rulemaking is a central part of our work to implement Executive Order 14300 which directs the NRC to review and modernize its regulations to support deployment of new technologies while maintaining our longstanding safety expectations. Our goal this morning is to give you a clear picture of selected portions of the rule that are of interest to the ACRS, how they work together and complement the overall rule,

and where your insights will be especially valuable as we move forward toward finalizing the changes in the proposed rule later this year. We can go -- thank you for the next slide.

The purpose of today's discussion is really straightforward. It is to provide the Committee an update on the portions of the proposed rule that intersect directly with your areas of review and oversight. We will outline how each component strengthens clarity, predictability, and the technical foundation of our regulatory framework. Go to Slide 3. Thank you.

Building on the ADVANCE Act, the Executive Order 14300 directs us to modernize our regulations in several key areas, including establishing clearer thresholds for design changes, construction, focusing safety assessments, and credible and realistic risk and reconsidering the regulatory structure for license renewal durations. We also took the opportunity to modernize some of our emergency preparedness requirements and look at durations of certain licenses amongst other activities to describe comprehensive rulemaking. It implements -- the rule itself implements those directives that I've mentioned and

brings forward elements of a broader wholesale revision and enabling additional power generation capacity in supporting the new reactor technologies deployment.

The rulemaking represents NRC's most comprehensive and modernization of the reactor licensing in decades, in particular, regulations Part 50 and 52 which are workhorse regulations for the agency over the last few decades. The rulemaking represents a full life cycle update spanning design, construction, operations, license renewal, and decommissioning informed by decades of regulatory experience and where the industry is heading.

The proposal actually contains 17 provisions in the proposed rulemaking package, each aimed at modernizing a different part of our framework. Taken together, they represent the most sweeping package of reforms the agency has put forward in these particular regulations in a very long time. The central theme is an increased shift to the approaches that are risk-informed and performance-based.

Risk-informed regulations focus on credible risk-significant scenarios and allows use of modern models, analytical tools, and current safety insights. Performance-based regulations focus on

outcomes rather than prescribed methods. Together, these approaches reduce unnecessary rigidity and allows us to regulate with precision while maintaining the safety expectations. Go to Slide 5, please.

Let me give you a brief walkthrough of the five major components that we're going to discuss today in more detail. Each of these addresses longstanding changes in our licensing framework. And together, they form a coherent modernization effort. The first topic is Appendix T, or proposed Appendix T, to streamline -- offer an alternative streamlined quality assurance approach.

This is a performance-based alternative to the traditional 10 CFR Appendix B, 10 CFR Part 50 Appendix B, quality assurance requirements and aligns with international standards and lets applicants to scale QA rigor to the actual safety significance of SSCs. It reduces unnecessary NRC oversight of suppliers while keeping accountability clear.

For many advanced designs, this provides a better fit than the current one-size-fits-all QA structure. It also opens up the supply chain market to enable access to a greater number of suppliers, both domestically and internationally, and offers more

flexibility than the current Appendix B approach.

The second topic is a consequence-based siting flexibilities that we're offering in changing the criteria. Second, we introduce a tiered, consequence-based siting approach that better reflects modern reactor designs, including low consequence systems. This expands viable siting options while maintaining dose-based protections, replaces older siting assumptions that don't align well with emerging technologies; in particular, for small modular reactors and microreactors. So we're encompassing a much broader range of reactor types.

The third topic we'll discuss today is determinate and data-based thresholds for safety assessments. We use the terms design basis events and beyond design basis events thresholds, or DBEs and BDBEs. Interesting enough, we have not actually in the past defined what a beyond design basis event is in terms of the regulations proper. And yet the regulations mention the use of these terms quite often.

We are proposing a use of frequency-based criteria, set of criteria, to focus safety assessments on events that are credible and risk-significant. It screens out extremely unlikely initiators and reduces

the churn we've seen applying traditional analysis methods across diverse designs. The aim is to result in more of a consistent predictable framework.

I'll note the Committee has vast experience, probably, observing the staff and applicants struggle through our Part 50 requirements, trying to make the regulations or the design meet those -- line up in those expectation. And, quite often, we spend a lot of time inefficiently debating about whether exemptions are needed.

The fourth topic is risk-informed, performance-based acceptance criteria. We've actually proposed a new pathway in Regulation 50.220 and a companion or conforming change in 52.220, in Part 50 and 52. This allows applicants to propose risk-informed alternatives in lieu of prescriptive acceptance criteria where again we're not relying on the exemption process, where regulating by exemptions has been an inefficient and historically challenging area. Given the timelines that we're trying to meet with more speed and be expeditious in regulatory reviews, both the applicant and the NRC staff are looking for more alternatives, and this provision give those alternatives to the stakeholders.

And then our fifth topic today is risk-informing the 50.59 regulation, provide more flexibilities for method changes, and also for using quantitative risk insights. We are proposing to enable the use of PRA quantitative risk insights to show that changes proposed for facilities would result in no more than a minimal increase in risk.

Historically, the regulation in 50.59 has not been updated in several decades. And during the last update there were restrictions placed by the Commission due to lack of maturity in some things like the probabilistic risk assessments and methods used for simulation modeling for safety analysis.

The second part of this change that we're proposing is for verification, validation, and quantification of uncertainties, which is VVUQ acronym in short, and allows for looking at programs and approving them beforehand so that licensees and applicants can use these approaches, vetted with the regulator, to go ahead and enable and adopt new analytical methods once the modeling basis is clearly established for the surrounding programs. It reflects a significant advancement -- both of these changes reflect significant advancements both in PRA modeling

over the last few decades as we've seen a lot of maturity in the PRAs, given the acceptability standards that the industry uses, and the peer review processes that the PRA models and the operating fleet have undergone.

And the modeling tools for simulation -- advanced simulation has matured quite a bit in both the nuclear sector and other safety-critical industries. So it was the time to refresh this regulation to go ahead and enable a more flexible framework.

Taken together with the other rulemaking proposes that I mentioned, in total, the 17 provisions of the rulemaking package, these components improve clarity, reduce unnecessary rigidity, and keep regulatory focus on credible risk-significant scenarios. They support innovation while maintaining the agency's safety mission.

With that said, I'm going to turn it over to the next presenter so that we can go into a more deeper discussion on each of these five topics. I believe Deanna is next.

MS. ZHANG: Hi. Thank you for that introduction. So, my name is Deanna Zhang, and I'm in the vendor and quality assurance branch. So my

focus -- the focus of my presentation today will be on an overview of 10 CFR Part 50, Appendix T but also on the changes that were made since the last presentation to the ACRS. This includes changes in the actual proposed rule language as well as changes in the FRN. Next slide, please.

So, as a result of concerns raised to the NRC, the NRC is proposing to add a new Appendix T to 10 CFR Part 50 as a voluntary alternative to Appendix B. The proposed Appendix T would draw on international quality standards and will incorporate elements for performance-based QA criteria. And it will explicitly include the use of a graded approach for applying QA requirements for SSCs relative to their safety and risk contributions to the overall nuclear facility.

There will be QA requirements included for software use in additional items and for design analysis. There will be new QA terminologies and methodologies introduced that will align with other safety critical industries and will leverage international standards for quality management. This will allow applicants to leverage across industry and global supply chains.

Lastly, the proposed Appendix T will

provide for increased licensing certainty by removing the need for NRC oversight of vendors of products and services related to nuclear power plants and fuel reprocessing plants subject to this proposed rule. And I will say for this last bullet that one of the changes that was made in the proposed rule was that the language is not as strong regarding the need for NRC oversight of suppliers instead of an affirmative that it will be removed. The wording has been changed to may as directed by the Commission such that in cases where you would need oversight of suppliers, it can still be done. And this will be based on any significant issues encountered during manufacturing or any other things that may come up such as allegations. Next slide, please.

MEMBER KIRCHNER: Will you take a question? This is Walt Kirchner.

MS. ZHANG: Yes.

MEMBER KIRCHNER: Yes. Since you brought that up, so then what is the trigger for such oversight of suppliers and vendors, a 10 CFR 21 defect?

MS. ZHANG: So we have -- in our vendor inspection program, we have a set of criteria for when we would do, essentially, reactive inspections. So,

for this proposed Appendix T, if it's used by the applicant, then what we can do is incorporate the criteria of focusing the inspections on more reactive inspections to -- for the suppliers that supply to applicants or licensees that are using the proposed Appendix T. And that's -- their criteria right now includes Part 21 defects, allegations, and other international issues that may be raised that have applicability to the U.S. market. Does that answer the question?

MEMBER KIRCHNER: Okay. Thank you.

MS. ZHANG: Next slide. So right now in the proposed Appendix T, there are eligibility criteria for use of Appendix T. So in order to reference this appendix in application for a construction permit, operating license, or combined license, an applicant must satisfy three criteria. So the application would need to be for an nth-of-a-kind construction permit operating license and combined license.

And that would need to identify the first-of-a-kind reference plant in the application and any departures from the first-of-a-kind plant. It will also limit the departures from the first-of-a-kind reference plant to those that do not change the

classification, design, and method of manufacture, construction, and operation of SSCs within the scope of this appendix. And, lastly, procedures and work processes for implementing the requirements in this appendix must be established and included in the application.

And I will say that one of the concerns raised during the subcommittee meeting was that this restriction for the eligibility to use Appendix T that it may force changes in how a first-of-a-kind plant would need to be transitioned to Appendix T when the first-of-a-kind plant was using Appendix B. And that may not be ideal and it may limit the use of Appendix T. So I'll address how we've addressed that concern in the next few slides. Next slide.

So key parts of the Appendix T include certain general requirements which are to establish a QA program, document the QA program in the Quality Management System and submit the Quality Management System for NRC review and approval. The Quality Management System must implement QA requirements in Section IV of the proposed Appendix T which includes criteria for management, performance, assessment, and for software used in design and analysis and digital

items important to safety. The implementation of the approved Quality Management System must be done.

And the QA program must cover safety-related SSCs and non-safety-related safety significant SSCs and also for the Quality Management System to document a graded approach on how the QA program will be implemented. And I just want to highlight that in the proposed rule, the one that is published in the FRN -- will be published in the FRN, we did make some conforming changes. One was to the definition of nth-of-a-kind.

That is to change the and statement regarding what is nth-of-a-kind to an or statement such that it will legally -- it will be more legally correct in what the staff intended for what is considered nth-of-a-kind. Indicia, we also made changes to the software used in analysis and digital items important to safety requirements to, one, be more aligned to the terminology used in the rest of the appendix so that it's not only focused on safety-related SSCs but to a broader set of SSCs to be covered. And also making some language improvements in that section. Next slide.

Okay. So we discuss essentially two

additional questions that are included in the FRN. So first is on the eligibility criteria and how can the NRC expand the applicability of Appendix T to first-of-a-kind plants and existing licensees. This includes guardrails or restrictions that may desirable.

So this will get some feedback on how to expand Appendix T to first-of-a-kind plants and also to allow existing licensees to use Appendix T. It also requests whether -- feedback on whether the NRC should consider alternative approaches to Appendix T such as endorsement of ISO 19443 or changes to Appendix B in lieu of publishing Appendix T. The second question is on the QMS change criteria.

Right now, all approved changes on the approved Quality Management System must be submitted under the proposed rule. So this question to the public would ask whether the change process should have a graded approach similar or mirror 10 CFR 50.54(a)(4) in which only changes that reduce the commitments be submitted to the NRC for prior approval. So --

CHAIR HALNON: Deanna, this is Greg Halnon. Before we go too much further, I want to circle back on -- you mentioned that you were addressing the

ACRS's concern earlier about multiple quality programs having to be established. And just to be clear, the concern was if a supplier was supplying to a first-of-a-kind at least at that point when we talked about it, it was going to have to be a quality -- I mean, an Appendix B program.

And if that same supplier was going to be supplying to nth-of-a-kind, it would have to be an Appendix T program. The question was why can't the -- well, the concern was is that they would be wasting a lot of time and money on having two different quality programs. And then further if somebody was building a first-of-a-kind and intended to go on to an nth-of-a-kind type fleet, likewise, they would be having to start with Appendix B and then change to Appendix T. Could you just kind of in a very clear way describe how you say you solve that? Or is that this question, first question on eligibility criteria you're still trying to solve that question?

MS. ZHANG: So that's what we're trying to solve was this first question which is to gather the essentially feedback from the public regarding how we can address that issue and any proposals that would essentially help address the issue you mentioned.

CHAIR HALNON: Okay. So you acknowledge the inefficiency of having two quality programs. You're still trying to work through what's the best way to solve that. Is that a fair summary?

MS. ZHANG: That is a fair summary.

CHAIR HALNON: Okay. Thank you.

MS. ZHANG: Any other questions?

(Simultaneous speaking.)

MS. ZHANG: Oh, go ahead.

MEMBER HARRINGTON: Just real quick, this is Craig Harrington. It's not as much a question. It's just a comment. The second eligibility criteria lists you haven't made any changes from first-of-a-kind to nth-of-a-kind. And, in particular, like, method of manufacturer, I could conceive of a case where you have a first-of-a-kind plant and the first few nth-of-a-kind where the method of manufacture of some critical component doesn't change. But then you figure out a better way to do it.

Does that then throw you back into an Appendix B program if you change the method of manufacture? So there's those kinds of practical complexities in having that kind of eligibility

criteria. So just a comment.

MEMBER MARTIN: This is Bob Martin. We're going to hear about the risk informing of 59 a little bit later. But looking at that -- really that same eligibility requirement, I can't help but feel like there's a natural tension with the statement there and the eligibility requirement about departures from first-of-a-kind.

And then, again, we'll hear later on risk informing. It might do the same thing as what you just said, Craig, basically to force somebody back into Appendix B. You make the value of Appendix T less attractive. As of course, Dave noted in his introduction, a lot of work has gone into this. But was there any opportunity really to collaborate with these other efforts to understand where they might create these tensions?

MS. ZHANG: So I think with the questions, it gives the public a chance not only to consider how to expand the eligibility to first-of-a-kind plants as well as to operating reactor licensees. It would also give them an opportunity to identify other issues with the proposed approach. So I think that would address your question.

MEMBER KIRCHNER: Greg, do you want to hold discussion until later? I mean --

CHAIR HALNON: I think let's keep -- this is Greg Halnon -- keep the questions to the slides for clarification; the deliberation we'll do later. And then if we have questions, we can call the staff and have them answer those questions during deliberations. But I'm going to make sure we get through the slides this morning.

So clarifying questions on the slide, that's great. If you get concerns and other things beyond that, let's get it in deliberation. Deanna, go ahead.

MS. ZHANG: Okay. So I guess, just to summarize, there are some changes that are made in the FRN, both in the proposed rule as well as to include questions for -- to seek explicit public feedback into areas. So that's everything I have. If there are no more questions, I will pass it back to -- or I will pass it to the next presenter.

MEMBER KIRCHNER: Greg, I think maybe it's timely, while we have Deanna on the line right now, to just make a general statement. If the intent is to build the first-of-a-kind to existing standards,

that's NQA-1 and Appendix B, and then nth-of-a-kind. And you're really talking -- we're not saying what we're really trying to do here which is expand the supply chain.

And that's what it appears Appendix T is really about, because Appendix B based on NQA-1 is a product safety standard. And ISO and Appendix T as you envisioned it is a process management standard and they're different. So if you want to retain the quality that we have achieved in the past in the first-of-a-kind, then NQA-1 is the way to go.

If you change the eligibility on this slide, it says, expand the applicability of Appendix T to first-of-a-kind, then that's a completely different discussion because then you're trying to put a process standard on a product, a safety quality standard. And there's a distinction. This has been noted by the staff in SECY 030117. There's quite a lot of effort has been expended on this in the past.

So we should be clear about intent. If the intent is to address supply chain issues for nth-of-a-kind, I think what you have is probably the right approach. But expanding that backwards to product quality I don't think works, i.e., the

first-of-a-kind, so the NQA-1, Appendix B kinds of standards.

CHAIR HALNON: It sounds like --

MEMBER KIRCHNER: Well, I'm trying to address, how can you expand the applicability of Appendix T to first-of-a-kind? Well, you can, but you will dilute the quality standards for the product in the first-of-a-kind. That's -- so if the intent is to address supply chain, the approach the staff has taken is quite good. But if you're going to work backwards, then that's a different set of questions.

CHAIR HALNON: Yeah, this is Greg. Let's let the question play out and let them resolve both public comments and the other comments. If you have a comment on the question itself, that's fine.

But we're not here to ask the question again or to say it's a good or bad question. Just let the public weigh in and we can re-engage if we need to on the answer to that question, if you will.

It is a pivotal question. Most of the questions in the FRN are very open-ended from the standpoint of looking for feedback. So we have several others like that, too. So let's -- during our deliberation, we can talk about the questions, if we

need to, about what our thoughts would be.

So let's go ahead and move on to the next topic. And then when we get into deliberation this afternoon, Deanna, if you're able to still stay on or we'll call you when we're about ready to talk about Appendix T. And that way, you can listen in and potentially provide some clarifying information for us.

MS. ZHANG: Will do.

CHAIR HALNON: Thank you.

MS. RADEL: Hello. My name is Tracy Radel. I'm a senior nuclear engineer in the Division of Advanced Reactor Engineering. And I will be presenting on the proposed rule and guidance for enhancing flexibility of reactor site criteria. Next slide.

This proposed rule repurposes Subpart A to Part 100 which is currently used for testing reactors as a middle tier between Part 57 and Part 100, Subpart B requirements, increasing regulatory flexibility. Subpart A would be applicable to Tier 1 reactors defined as power reactors having an unmitigated consequence of less than 25 rem, total effective dose equivalent, or TEDE, at the site exclusion area boundary, or EAB.

Power reactors with an unmitigated consequence of greater than 25 rem TEDE at the site EAB or where the unmitigated consequence is undetermined would remain under Part 100 Subpart B.

Changes are also being proposed to 10 CFR 100.11 to be more technology inclusive and aligned with other areas of regulation. Next slide. The related guidance with this proposed rule includes a revision to Reg Guide 1.208 which provides a graded approach for defining site specific earthquake ground motions and endorses ANC ANS 2.27 2020, ANC ANS 2.29 2020, ASCE/SEI 43-19 with clarifications. Also included is our revision to Reg Guide 1.23 which provides a new appendix on use of offsite meteorological data for nuclear facilities, providing guidance of aspects such as how to ensure data quality and that the data is representative of the site.

And, finally, Draft Guide 4036 provides a graded approach to site characterization for all hazards. The next slide here provides additional information on DG-4036 which contains the bulk of the new guidance supporting the proposed rule.

Next slide. DG-4036 includes guidance on graded approaches for geology, seismology,

geotechnical engineering, hydrology, meteorology, and human induced external hazards. It discusses the use of existing site characterization information as well as alternative site investigation methods for the different hazards and provides approaches that can be scaled based on characteristics and risks associated with the particular reactor facility and the site. It also includes guidance -- DG-4036 also includes guidance on how to perform an unmitigated consequence analysis for determining the reactor tier and a nuclear site and category which is used as a metric for graded approaches for some hazards. DG-4036 also provides guidance on aspects to be considered when developing site parameter envelopes.

And lastly, it provides an appendix with methods for optimizing the seismic design basis for nuclear facilities to a consideration of key aspects related to siting and through the use of multiple seismic design categories or SDC levels. This is my last slide. And I will take any questions from the members now.

MEMBER ROBERTS: Yeah, Tracy. This is Tom Roberts. Can you speak to the process for selecting the accidents that will be considered the

unmitigated consequence analysis? For example, would include it include reactivity addition events in addition to, say, loss of power?

MS. RADEL: The reactivity addition events would be based on credible scenarios that could be caused by the external hazards. So things like compaction of core due to seismic. But it would not include all potential reactivity insertion. It would be based on what is reasonable for external hazard -- accidents induced by external hazards.

MEMBER ROBERTS: Thank you. So the intent is to consider accidents that could be caused by the external event, not necessarily all hazards that could be unmitigated by systems that could be affected by the external event. Okay. I think I understand. Thank you.

MEMBER HARRINGTON: Tracy, this is Craig Harrington. I apologize. I just don't remember what it says in here in the guidance. Is there some element that addresses multi-unit sites and how you treat that?

MS. RADEL: Yes, there is specific guidance within DG-4036 on how to handle multi-unit sites or sites with different types of facilities that contain radiological material.

MEMBER HARRINGTON: Okay, good. Thank you.

MS. RADEL: Any other questions?

CHAIR HALNON: I don't see anymore here, Tracy.

MS. RADEL: Okay. Thank you.

MS. TULIP: Good morning, everyone. I'm Jacqueline Tulip. I'm a project manager in the Division of Advanced Reactor Licensing. I'll be discussing the portion of the proposed rulemaking associated with the use of data-backed and determinate thresholds for reactor safety assessments. Next slide, please.

In approaching this proposed rulemaking, the staff recognized that a significant portion of reactor safety assessments are done to assess the plant response to design basis events or DBEs historically using methods consistent with NUREG-0800 or the Standard Review Plan, SRP, Chapter 15 on transient and accident analysis. The guidance related to determining the spectrum of DBEs in the SRP Chapter 15 is qualitative and was developed from judgment and operating experience with large light-water reactors. The DBE spectrum and the SRP does not reflect the use

of a determinant and data-backed threshold and does not always align well with evolutionary light-water reactor or advanced reactor designs.

In response, the staff developed the proposed rulemaking to introduce the definition of DBE to 10 CFR Part 50 that enables the use of thresholds in an effort to better support evolutionary light-water reactor and advanced reactor design and licensing processes by allowing removal of less frequent initiators from reactor safety assessments. Next slide, please. The proposed rulemaking includes provision of 10 CFR 50.2, definitions, to include definitions of DBE as well as beyond design basis event or BDBE. These definitions will establish that categorization as a DBE or BDBE is based on initiating event frequency thresholds.

The staff focused on initiating event frequency rather than something like event sequence frequency or consequences to enable applicants and licensees to use more readily available data and to avoid the approach requiring a detailed probabilistic risk assessment. There are already other frameworks that take event sequence and consequence factors into account which applicants can pursue if desired.

Proposed definitions are intended to establish the following.

The DBE category will capture critical events whose treatment will be consistent with current regulatory practice. The BDBE category will capture credible events that fall below the DBE threshold using more realistic methods and will establish a frequency below which no regulatory treatment is necessary. Regarding the proposed BDBE definition, I'll acknowledge that 10 CFR Part 50 includes specific provisions for a number of what could be considered events such as station blackout or anticipate transient without scram.

This proposed BDBE definition does not eliminate any requirements to address events specifically required by the regulations. Next slide, please. Here you'll see numerical thresholds proposed to be used for categorization of events. Those are greater than or equal to 10 to the -4 per year for DBEs and less than that but greater than or equal to 10 to -6 per year for BDBEs.

These numerical thresholds themselves are not explicitly included in the proposed rule language and are only included in the associated guidance to

allow for flexibility of applicants and licensees to propose unique thresholds with supporting justification if desired. You'll also see the types of treatment for each of these event categories. So that's traditional conservative methods such as those outlined in SRP Chapter 15 for DBEs, assumption of realistic conditions, and the use of best estimate methods for BDBEs and no regulatory treatment for events below the threshold for BDBE. Next slide, please.

The staff has drafted Draft Regulatory Guide 1454 to implement proposed definitions and the categorization process. The guidance includes those numerical thresholds and graded methods discussed in the previous slide, qualitative criteria for categorization as well as brief information on the use of thresholds to inform design bases parameter selection. The steps you want to emphasize, the inclusion of those qualitative methods is in the draft guidance.

This is not solely a numerical approach. For example, qualitative criteria for inclusion in the spectrum of DBEs outline in the draft guidance includes any single operator control input that could

cause a plant perturbation and lead to an undesirable plant state or any single control system failure, including failure of the final actuating device that can cause a plant perturbation and lead to an undesirable plant state. The staff also want to note that the draft guidance does include criteria and a procedure for in limited cases combination of an event initiator with one additional independent failure of a non-safety related SSC to define a single design basis event.

This is a case that applicants have proposed and the NRC staff have encountered in previous advanced reactor licensing experience. So the staff hope to reduce ambiguity and streamline decision making by outlining those criteria and procedures for future cases of this in the draft guidance.

Next slide, please. For applicability, the proposed definitions are intended to be a voluntary option for current commercial power reactor licensees and current applicants.

Proposed definitions are intended to be mandatory in keeping with current practice for definitions under 10 CFR Part 50 for future commercial power reactor applicants for applications submitted

on or after six months after the effective rule date. The expected impact is to better focus reactor safety assessments on credible realistic risks rather than relying on prescriptive spectra of events within the existing Part 50 and Part 52 frameworks by improving clarity and regulatory certainty for applicants and for NRC staff.

As Mike Franovich mentioned, this approach is expected to streamline licensing decisions and reduce the need for applicants and licensees to request exemptions. I believe that is the end of the section.

MEMBER MARTIN: This is Bob Martin. A clarifying question regarding how this change might change how a license holder or applicant would approach ATWS and SBO where they might be able to show through their PRA, these are maybe beyond 10 to the -6, something like that. And then that would be the argument strictly based on a PRA. Or would you still expect it'd be a traditional analyses that go along with those sections, Part 50. Is it 62, 63? Is there any specific impact that you might expect a change like this?

MS. TULIP: So the change is not modifying any existing requirements for those events like you

mentioned.

MEMBER MARTIN: I guess I was concerned that it might make for a very easy way to get through 50.62, for instance, but just showing through the PRA numbers that this falls below a credible definition. And we'll see utilities, applicants depart from the kind of safety assessments that have done in the past strictly through risk and argument. Those parts were put in there as kind of deterministic activities dealing with the high uncertainties of those kind of events.

Clearly we know more today. So maybe there are arguments for change. But it's just going to become a checkbox. That was kind of the question should the PRA show that these are incredible type events.

MS. TULIP: I think it's possible that the PRA as you mentioned or an analysis could show a frequency less than the incredible thresholds. However, there's nothing in the proposed rulemaking that would eliminate the need to still address those events. Perhaps that analysis could form the basis for an exemption from requirements to address specific events. But this proposed rulemaking doesn't remove

the requirements to address those events.

MEMBER MARTIN: Thank you.

MEMBER PALMTAG: This is Scott Palmtag. I had a general question on some of the current commercial power reactors. So how exactly will these criteria, the one -- even the one of six selected? So if a current commercial power reactor did volunteer this new definition, would you expect there to be any reclassification of current events, for example, current events that are beyond design basis -- or I'm sorry, DBEs being classified to BDBEs, if I can say that right? Or is this 1e to the -6 line up with current industry practices? I'm kind of curious if you've looked at how this will impact the current commercial reactors.

MS. TULIP: We did pull from a number of different places when we were considering these thresholds to include in the guidance. I believe the Level 3 PRA project was one of those factors. And I apologize for putting him on the spot.

But if Kevin Coyne is in the room, I believe he did run a test case with some of these thresholds. Is Kevin there? Is that something you can speak to?

MR. COYNE: I'm here, Jackie. So, yes,

we looked at -- we did have insights from the Level 3 PRA project that is wrapping up. We also looked at the operating experience data that INL collects. So they collect initiating event data probably since the beginning of the industry. It really goes back that far.

So we looked at the spectrum of events that we would consider initiators based on that data versus the typical things that are considered. So for the operating fleet, things that get pushed out are kind of rare events. And we have Chris here, too, who can speak to this.

But rod ejection type events probably would be ones that would go into the DBE or lower category. The other typical events that are in the spectrum in Chapter 15 generally wouldn't change. The one exception is large break LOCA, and that's handled by 50.46.

So these definitions wouldn't change the treatment under 50.46. The interesting one that comes up when you ask for determinate, data-backed thresholds, the operating experience leads the way. So, electrical faults, we have plenty of data for. And they're not infrequent events. They're

1E-minus-3 type events for loss of like a Class 1E electrical bus. So those would be ones that we would expect to be now in the spectrum of events that could be considered. So I think dealing with those types of events would be difficult for an operating plant. For a plant that relies on more passive equipment, that probably is a less significant event to deal with.

MEMBER MARTIN: Okay. Thank you.

MEMBER HARRINGTON: This is Craig Harrington. Just as a practical implementation piece, if an existing plant voluntarily chose to implement this, would you expect them to make a wholesale change or pick and choose when they apply or don't How, as practical matter, would they implement it? That make sense?

MS. TULIP: That does make sense. And the intent is that the approach would be adopted wholesale because if the goal is to better focus on credible realistic risk, it should be approached holistically. If the numbers show that there's a credible risk that exists somewhere that maybe wasn't considered before, we would still want that to be included in the approach. So to answer your question, the intent is for it to be adopted wholesale.

MEMBER HARRINGTON: So it would be somewhat of a licensing project to work through the process and declare it implemented for their plant. And then from that point forward, that's the way they would behave. Okay. Thank you.

MEMBER ROBERTS: This is Tom Roberts. I was going to follow up on Bob Martin's comment on ATWS. ATWS is kind of an odd one because it's dated as a deterministic requirement in the regulation. But it's based as a 10 to the minus-5 frequency. And so if somebody who wasn't covered by that regulation which is basically any plant that existed in 1983 were to meet the ATWS requirement, they would use the 10 to the minus-5 threshold presumably which seems potentially less conservative than the 10 to the minus-6 that's now being put out in this guide as the boundary of a beyond design basis event.

So I guess it might be a question for Kevin again since you sat back down there. Have you thought about that? Have you thought about whether this actually widens the aperture for the types of scenarios that need to be assessed under ATWS for plans not covered specifically by the regulation?

MR. COYNE: So that's a great example.

And actually station blackout when the rule was formulated was based on risk insights that existed at the time that obviously evolve with time. The 1E to minus-5 is a pretty common criteria that applicants come in and request exemptions to ATWS from.

It isn't a purely numeric evaluation, though. We'll look at the PRA results that substantiate the 1E to minus-5 for this scenario and work through the exemption criteria. But for the sake of streamlining this rulemaking, we really didn't want to touch the other regulatory requirements.

It would be very case specific as to whether it should be changed by the definition or not. The way we've worded the actual regulation, applicants have a great deal of flexibility in how they determine the definition. And they could certainly propose definitions that eliminate those events. And we would look at them on their merits when we review the definition.

MEMBER ROBERTS: So you would look at the 10 to the minus-5, or 10 to the minus-6 in the case of an actual application, as opposed to predetermining now that you're changing that?

MR. COYNE: Correct. And then the option

always would be available for an applicant to come in and request an exemption to the ATWS requirements as some have done in the past.

MEMBER ROBERTS: Right. So it seemed kind of odd for an event that would screen as a BDBE to be screened out because of a deterministic requirement for ATWS. It just seems like a strange place to be.

MR. COYNE: So yeah, again, it becomes a very case specific type of review. And Jackie didn't cover the actual definition. But it's a very applicant flexible approach that they can tell us what their definitions would be and what the thresholds would be and how they're applying qualitative criteria, those types of things. So we would have to take a look at that. The guidance would have you use these thresholds and additional qualitative guidance.

MEMBER ROBERTS: So the exceptions like ATWS were intended to make it less restrictive?

MR. COYNE: Right.

MEMBER ROBERTS: Because it sounds like it's more restrictive because you have to look at ATWS regardless of frequency.

MR. COYNE: Correct.

MEMBER ROBERTS: But you can also interpret that as you actually got an order of magnitude less you have to worry about using a historic basis for ATWS.

MR. COYNE: So the draft guide goes through some discussion on the basis for this. But the definition itself talks about any event required by regulation -- by other regulation could still be in the mix. So you couldn't use the definition to get rid of those.

Otherwise, you have to consider them and whether we do an exemption with the definition coming in or whether we could do a self-directed exemption from the staff side. We'd have to look at the details of that. But that is a fairly common one that the new applicants come in with looking for relief on.

The other one that comes up that we're keeping just to be complete as a design basis event is the design basis accident analysis which is the substantial release of fission products into containment. So we're not looking at what the frequency of that type of event would be. It's a deterministic analysis done to assess the functionality of containment and other engineered

safeguards. So that would still remain as a DBE regardless of the definition you came up with. So that would be the intent is to keep that in the DBE category even if the accidents that would result, it would be a lower frequency.

MEMBER ROBERTS: Right, yeah. Thank you.

CHAIR HALNON: Looks like there's no more questions on this one.

MR. HUGHEY: To avoid confusion -- just to avoid confusion, so no one thinks I'm Jacqueline. Good morning. I'm going to talking today. You can go to the next slide, please.

I'll be speaking this morning about the proposed 50.220 and 52.220 proposed regulation. So these proposed changes would allow licenses and new reactor applicants to request alternatives to acceptance criteria across all of 10 CFR Part 50 and Part 52. Alternatives would be requested using license amendment request as opposed to an exemption which is the traditional method.

So this approach provides efficiency gains because it removes the requirement associated with an exemption that demonstrates special circumstances. So the special exemption regulation, 50.12, includes

six special circumstances, one of which must be met to even consider what alternative is being proposed. So these special circumstances are things like the change continues to meet the underlying purpose of the regulation or even compliance with the regulation and the circumstance presents an undue hardship to the licensee while not complying still presents no undue risk to the public health and safety, those sorts of things.

So the situation that we find ourselves in is that so in the case of new reactors, as you're aware, new reactors use extensive use of passive safety systems as opposed to active safety systems. And they also -- they have new technologies that are being implemented whereas the Part 50 and Part 52 were developed in the context of having active systems for safety that provide safety functions. And they were large light water reactors whereas many of the new reactor designs are non-water reactors, especially small modular reactor designs.

And so what happens is when we get in an application for a new reactor, often there are exemptions proposed or discussions about whether or not an exemption needs to be proposed. And there are

extensive discussions about, well, how does -- what's the special circumstance? How does that align?

How does this new technology, this new design align with one of these six special circumstances that needs to be satisfied in order to even propose an alternative? So that can create ambiguity and complications in the right. We see this also in existing reactors.

I've had many occasions where an exemption is submitted to us. And the technical adequacy of the exemption itself is not in question. It's straightforward.

But it's difficult to actually match up one of the six special circumstances. And so this complicates that review. So this approach -- in this approach, no overarching or compelling reason is required solely to allow a technical alternative to be considered. And so in this approach, an alternative can be evaluated based on its own merits without first meeting a threshold that's often unrelated to the technical adequacy of the proposal.

So that, in essence, lowers the entry threshold for consideration as compared to an exemption. However, while that threshold is lower,

the approval standard under the proposed regulation is somewhat higher. So in the proposed standard, the alternative must provide the same level of safety compared to whatever acceptance criteria is being changed out.

And the approval standard for an exemption is that there be no undue risk to public health and safety. So this has a higher standard of acceptance. So also I want to mention that in the proposal also only applies to acceptance criteria within the regulations.

So let me say something about that. So of course there are numeric acceptance criteria obviously. But there are lots of other acceptance criteria throughout the regulations that deal with how you -- the actions that need to be taken, steps that need to be taken, methods that need to be used in order to meet the regulation itself.

And so there's lots of areas in which acceptance criteria would be available for licensees to provide alternatives to other just numerical acceptance criteria. So these kinds of guardrails that are structural to the regulation, they're there to prevent requests that would effectively restructure

-- have a request to restructure an entire regulation for an individual licensee or an applicant. So another efficiency gain is that the regulation itself specifies what must be addressed in the technical justification.

And so the kind of elements it specifies are considering the results or insights from a probabilistic risk assessment or another systematic risk assessment which could be qualitative or quantitative. And also ensuring that defense in depth is maintained, safety margins are maintained. And it also requires that the use of an integrated panel that's made up of plant knowledgeable staff to review and confirm the appropriateness and acceptability of the proposed alternative.

So this provides consistency and clarity regarding the scope and standards needed to support an acceptable justification that demonstrates an equivalent level of safety for the proposed alternative. So these elements are incorporated into the proposed rulemaking to enhance regulatory flexibility, provide clarity in how to justify and propose a justification for an alternative and also maintain appropriate regulatory oversight. So finally, I'll just mention also there is a small

revision to 10 CFR Part 50 Appendix A, the general design criteria appendix in the introduction that just makes an explicit statement that the alternatives associated with the general design criterial may be justified within licensing submittals without the need for an exemption.

I think one over thing I should probably mention, it's not listed on the slide there. But there is also an FRN question that's been proposed. I believe it's Question 41. And what that question is it's looking to explore ways to leverage beyond what is currently listed in the 50.220 regulations.

And that would be -- that's requesting input, any comments, ideas from the public regarding how the NRC might going forward be able to have a repository of approved alternatives that may be generically applicable to plants and that plants would be able to implement without necessarily requesting a license amendment request and yet still be consistent with requirements of 50.59. So that's a question that's out there as well. And I just have the one slide, so I'm happy to answer any questions.

MEMBER KIRCHNER: This is Walt Kirchner. John, what would you expect just teeing off the last

bullet there in such a licensing submittal as technical justification for changing any of these criteria and/or the GDCs? Are you looking -- are you expecting there would be a substantial technical justification behind what's proposed? Of course it depends on which criterion --

MR. HUGHEY: Right.

MEMBER KIRCHNER: -- you're trying to change. But where I'm going with this is, for example, we've seen submittals where people say we're using functional containment. And they reference the SECY on functional containment, but there's no technical justification to substantiate that a functional containment approach would really work for that particular concept. So it's kind of -- would you -- for something like that under 50.52, I would expect a substantive technical justification to back up a change in one of these criteria.

MR. HUGHEY: Yes, that's correct. And that's why it's important that the elements for that technical justification are actually listed in the regulation. So with an exemption now, you don't have that necessarily. With this proposal, you do.

You have specific elements that must be

present in the technical justification. And so this is really a partial integration of the exemption regulation 50.12 and the 50.59, 50.69 categorization -- risk-informed categorization of SSCs. And the portion it's pulling from, 50.69, is really those required elements to provide the technical justification.

MEMBER KIRCHNER: Thank you.

MR. HUGHEY: Sure.

CHAIR HALNON: John, this is Greg Halnon. I need to get up and leave in a few minutes. But I just wanted to pose this and get into your head a little bit on the higher threshold approval for same level of safety versus undue risk. Both of those in many cases are judgment calls based on experience of the staff. Can you just give me an example? I mean, what I don't understand is when it's a numerical issue, it's not a problem.

MR. HUGHEY: Right.

CHAIR HALNON: I get it. But why do you think it's different? What do you think that there's one lower level standard versus a higher level standard there? Isn't the whole aspect is it safe enough?

MR. HUGHEY: Well, okay. So you're

asking why is the standard different, not is the standard really different?

CHAIR HALNON: Well, no. I'm asking why do you think it's a higher level -- higher safety threshold when you're saying same level of safety versus no undue risk?

MR. HUGHEY: Oh, sure. Well, because with a traditional exemption, the requirement is not whatever I'm doing now meets exactly the same level of safety as what I'm replacing. And I think the most dramatic example of that would be a hardship special circumstance. So a licensee comes in and says, I'm just not going to do this because it's an undue hardship.

These parts are obsolete. I can't replace them. But we don't even credit this system in our PRA from accident mitigation anymore. We've worked around it. We got other things.

And so for that reason, we've got adequate protection. There's going to be no undue risk. Well, not doing that if they have that system in addition to all those other things that they had, that would still be -- you would still have a higher level of safety.

You're completely taking that out of the way so it's not even there now. And then those are varying levels. So you're right. But I don't dispute. You're right.

It is a judgment call. With any regulation that you develop, you're looking to balance two things, flexibility and clarity. And if you have a regulation that's 100 percent clear, no judgment required, no judgment, it's 100 percent clear, you usually don't have very much flexibility.

And if you have a regulation that's very flexible, lots of leeway, well, then you don't have a lot of clarity normally. So the idea is to balance those two things. And I think that this proposed regulation using integrating the elements as it does, does a very good job of kind of balancing that flexibility and clarity together.

CHAIR HALNON: Okay. I think I get where you're going. And I guess also when you talk about technology inclusive, it's essential because some of these may not just simply not apply --

MR. HUGHEY: Correct.

CHAIR HALNON: -- to that technology.

MR. HUGHEY: That's right.

CHAIR HALNON: Okay. Thank you. At this point, I'm going to turn the meeting over to Craig Harrington as acting chair. I've got a meeting to go to. So carry on.

MEMBER HARRINGTON: No additional questions, I think. So, okay. Move on to the next topic.

MR. HELLER: All right. My name is Kevin Heller. I'm a nuclear engineer and technical reviewer in the nuclear engineering branch in the Division of Risk Assessment and Safety Systems. I'm here today with my colleague, Charles Moulton, who's joining us remotely. And we're going to be speaking on risk informing 10 CFR 50.59 and allowing flexibility for changes in methods.

Next slide, please. A good place to start, I think, is with just a little bit of background. Not like it's any surprise to anybody.

The intent of 10 CFR 50.59 is to allow licenses to make changes to the facility without a licensing action provided acceptable levels of safety are maintained as documented in the final safety analysis report. And there are -- I like to refer to them as constraints. There are two constraints

associated with this that we're really touching on today.

First one being that in the 1999 statements of consideration, it established that risk results would not be appropriate to evaluate a change to the facility under 10 CFR 50.59 without further rulemaking. Then the phrase departure from a method of evaluation within 50.59 was narrowly defined to restrict the ability to make non-conservative changes to analytical margin without NRC review. And so there's two proposed changes that are associated with this rule then or this rulemaking rather, one for each of these constraints.

The first one would allow use of quantitative results in risk-informed probabilistic risk assessment based evaluations to show that an increase is not more than minimal under 10 CFR 50.59. And then there's a proposed additional exception to 10 CFR 50.59 permitting method of evaluation departures when they're supported by a documented verification, validation, and on certain quantification process when it's performed under one of those that is 10 CFR 50.221-compliant. So let's dive into the first one then and I'll pass it off to Charles to speak on that. Next slide, please.

MR. MOULTON: This is Charles Moulton. I'm in the risk assessment and fire protection branch in the same division as Kevin. So when we say we're intending to risk inform the screening criteria in 50.59, we intend to risk inform the first two criteria in Section (c)(2).

And also the use of quantitative risk results is permitted only if the defense-in-depth and safety margins are maintained and documented. Also the PRA must cover all relevant internal and external hazards. The hazards is excluded.

The exclusion must be justified and documented. The licensee using this approach must also identify the NRC-approved applications that address the adequacy of each of the relevant hazard models being used. If they can't document where a hazard model has been used in an approved application, they can't use quantitative risk results to make this judgment.

They also can't use results if they're open facts and observations relevant to the change. To go along with this to ensure ourselves that PRA being used is of adequate -- technical adequacy, there's a number of things that need to be documented, including any

screening that was done, circuits that were used, and justification for decisions that were made, also documenting the uncertainties, the assumptions, any model updates and peer reviews that were related to change since the application that looked at the model. All this is fleshed out in the draft revision to Reg Guide 1.187 which also includes the proposed thresholds.

We're proposing thresholds of a delta CDF of 5 times 10 to the minus-7 per year for CDF and 5 times 10 to the minus-8 per year for delta LERF. And also a requirement that the baseline CDF and LERF must remain below Reg Guide 1.174 cumulative thresholds of 10 to the minus-4 and 10 to the minus-5 after the change. Next up is Kevin again about VVUQ.

MR. HELLER: Thanks, Charles. So when we talk about the second proposed change, the inclusion of additional exception to permissible departures from a method of evaluation, what we're really talking about is the new proposed section here, 221. This provides a regulatory pathway for licensees to change methods of evaluation under 50.59 when the change is supported by an NRC approved VVUQ program.

And the requirements of that program are

discussed within this section. From a higher level, we could say that really what this section is doing is it shifts licensing oversight from reviewing results, specific results, to reviewing the credibility basis for the models that generate those results. And so there are really three core concepts that are defined within this section in order to achieve this.

So the first is credibility. Credibility is defined as whether a model can actually be trusted for a specific purposes. The second is VVUQ which is verification, validation, uncertainty, quantification, and all other activities used to establish the aforementioned credibility.

And then there's modeling and simulation risk. This is the combination of the likelihood that the modeling and simulation is incorrect or misleading and the consequences of relying on the incorrect or misleading result for some decision making purpose. So with those concepts defined within the section, it then discusses categorically two requirements that any VVUQ program must include.

And the first of those are VVUQ activities. These are specific VVUQ tools you would actually be

using within the program, so for example, code verification or solution verification. The second then is VVUQ assessments.

And that's really what are you doing with those results and how are you assessing those results when you've utilized your VVUQ tools. So this might sound a little new or maybe -- dare I say maybe slightly radical. But the reality is the NRC staff over the years has approved well defined narrowly constrained change processes as part of various topical reports.

And so really that's what this is. And so I really like the final sentence here on this slide because what we're really looking at is that this is not blazing a new trail. This is stuff that we've seen and done before. This is more like turning a well-worn path into a road. And I think that's really important to emphasize.

So, with that, I'll take any questions.

MEMBER MARTIN: This is Bob Martin. I think there's a unique challenge coming to any kind of safety review of computer codes moving forward in the age of AI where -- again, I've been playing around with programming on this. And the tools that people are using -- and they have to, right, just to be

competitive with their own companies.

They create a lot of crap, a lot of fat let's just say, to the point where it can be difficult, almost impossible to read these codes because they're not elegant. And I'm focusing in on the first V. But it has an effect on the second V in particular because what I've observed is that the tools that are being used rely heavy on conditionals, right?

I can solve the problem with if/then blocks and what have you. If I try Solution A and it doesn't quite work the way I intended, it doesn't -- the tools don't necessarily erase what it did and try again. It leaves that code in -- those subroutines, those functions, whatever, it leaves that in there and then it works around it.

And you might test something and whatever domain that you're trying to solve. But that fat, you don't know necessarily whether it has created a bifurcation in the solution. And we're taking the human out of this sort of stuff.

And then at the same time, we're creating a -- it isn't coincidental. We're creating a risk informing of a world that we have a different mindset on because it was built on humans developing these

tools, not machines or software. And I'm not sure we have the data to really be comfortable with risk informing a tool that is now being or may be developed with AI.

I'm not sure it's the right time for changes like that. If we are having a whole different paradigm switch with how these tools are being created. And granted, about 99 percent of everyone that's coming is coming with a tool that was created by a human.

But, long-term, I think there needs to be some kind of check that can -- maybe it's an AI tool or whatever to kind of look at this unique challenge I think is coming to programming and what have you that the more you incorporate particularly these kind of conditionals and codes, it just creates more and more opportunity for error. Now granted if you have well trained people preparing these tools and well trained analyst should be able to clean these up. But I don't know whether focusing in with risk models on -- a risk argument on changes in light of tools, I think, going forward will be developed is necessarily enough.

I mean, this is my background, right? So when we, of course, would have a change, yes, it was a major effort, very expensive. Oftentimes, a year,

year and a half, several FTEs involved, expensive. But you ended up doing your full suite of analyses, unnecessarily risk informing out chunks of it. Okay. This was painful and I hated it. And my boss hated it.

I mean, it's a real challenge and it's good to be searching for solutions. We trusted it because we were -- we knew what was in there. I look at the future going forward, and people won't know necessarily what's in there.

Anyway, it's a reason to be a little anxious and not too confident that risk informing these verification aspects is enough. I don't know. I mean, I'm not going to say completely discourages activity because, again, I lived it and it's painful.

But, at the same time, it is something we need to kind of keep in the forefront that how codes are being developed today is different and we're losing the human in the loop that really is essential in that verification part. Validation, we have to test and I guess oftentimes a solution to with verification it is more extensive validation and uncertainty qualification.

But we cannot lose -- we can't diminish

that first V, because it's a risk argument. Because I'm not sure the world today is the same as the world of yesterday that we're applying the data from moving forward. So you got bad data, then you might make bad decisions. It's a little bit of a rant.

But I do think the timing of this is interesting because we're having a complete paradigm shift in how we approach code development. Not really a question but it's a reason to be a little tentative about approaching this challenge. Maybe a little philosophical.

MR. HELLER: Well, I don't want to downplay the concern. Something I would point out, though, is that we have to remind ourselves that where we are at the -- and when I say we, I don't necessarily mean the NRC. I mean all of us, the industry, the NRC, everyone.

When it comes to artificial intelligence, it's still in its infancy. It really is. And so with regard to the development of advanced codes using some kind of artificial intelligence, it does at the moment produce not the cleanest, concise-est code.

And certainly, in those instances, we should have some additional caution when taking a look

at those kinds of methods. That would be my personal position on that because it has removed a human component.

On the other hand, there are still a lot of methods that are being generated purely through human hand if you will that are very well -- that it very well assessed and that could utilize the proposed changes that are being made here. And so really I think it's just a challenge for us as an agency moving forward as to being aware of that. And I mean, I could speak at length on this. But for the sake of being concise, I won't say more than that.

MEMBER MARTIN: Yeah, if it came up a few years ago, wouldn't even think twice. This sounds good. But the world changes. And that's a challenge with risk informing anything. As the work changes, the database effectively changes.

MR. HELLER: Right.

MEMBER MARTIN: And, consequently, the uncertainties go up.

MR. HELLER: And, truthfully, what we're seeing here is, to a very real extent, an evolution of our regulatory structure and response to the increased computing power that has been achieved in

just the last 20 years. Maybe we need to reassess again in another 10. Who's to say once AI is more advanced.

MEMBER HARRINGTON: And this is Craig Harrington. You also have the opportunity because the VVUQ program would be NRC approved --

MR. HELLER: Exactly.

MEMBER HARRINGTON: -- to make sure that there are elements in that program that acknowledge this concern and try to manage it in some appropriate way.

MR. HELLER: Exactly.

MEMBER HARRINGTON: Scott?

MEMBER PALMTAG: Yeah, this is Scott Palmtag. I have a question not related to AI. So I have a lot of experience with the VVUQ tools. And my question is really not sure -- quite sure how this is going to work and if there's any experience in it.

But I'm just kind of thinking through this. You make a code modification. It can be a very minor code modification. You roll it up. It's going to change your results at some point.

These results are going to have to roll up into a safety analysis. You're going to have to do your UQ on it. You're going to have to take that

safety analysis and you're going to have to roll that up into your PRA and then into your CDF.

It's going to take a very big code change to change your CDF in any meaningful way. It just seems like there's a lot of code changes that can be hidden in here. So do you have any experience on actually doing this where you've made a code change and rolled it all the way up through all the different UQs and all the different PRAs and actually seen a change in the CDF?

MR. HELLER: So, Charles, I think I'm going to hand that one off to you simply because I don't frequently find myself involved with core damage frequency assessments. What I can say is --

MEMBER PALMTAG: It's a long way from a code change --

MR. HELLER: Exactly.

MEMBER PALMTAG: -- to the CDF.

MR. HELLER: So as I'm hearing your question, it's have I, in my experience, seen something that has actually resulted in a change to a CDF?

MEMBER PALMTAG: What's driving this? Do you have some examples of something where this has been used before and shown to be useful?

MR. HELLER: Oh, the inclusion of some kind of VVUQ program you mean?

MEMBER PALMTAG: Yeah.

MR. HELLER: Yes, there actually have been -- in fact, recently, there was a review that was done where the staff assessed whether or not certain well defined changes to a model, an empirically derived model would be acceptable when there was the inclusion of new data.

MEMBER PALMTAG: I know it's probably proprietary.

MR. HELLER: It is.

MEMBER PALMTAG: Can you just give us a rough idea of what this program looks like or what the code change was and how it rolled up and changed the CDF?

MR. HELLER: Well, I can't speak to how it would roll up and change a CDF because, again, that aspect of it, I haven't seen. But as far as the model itself and the changes, what was being discussed was the inclusion of new data. There was the restriction proposed by the applicant that the form of the model would not change.

So, for example, I'll make something up.

You wouldn't be going from, say, a second order polynomial to a 36 order polynomial. That would not be allowed. But with the inclusion of new data, you could with the existing form of it refit the model with the additional data. And that might allow for increased accuracy of the prediction results. But the results of then the inclusion --

MEMBER PALMTAG: I'm sorry. I find it hard to believe to that going from a second order model to a -- a second order polynomial to a 36 order polynomial is going to move your CDF value. That's why I'm just kind of looking -- I'm trying to understand how this would --

MEMBER HARRINGTON: The point wasn't -- this is Craig. The point wasn't that kind of a change would change your CDF. It was that that kind of a change wouldn't be allowed without NRC involvement.

MEMBER SUNSERI: Well, what I hear Scott saying -- and this is Matt Sunseri -- is that if there's a change to the code that results in outputs that are so dramatic that they affect the CDF, then how good was the original code?

MEMBER PALMTAG: Yeah. I'm just trying to understand an example where this would be used.

I'll just leave it as a comment.

VICE CHAIR PETTI: So this is Dave. My sense is this may, in fact, shed a lot of light on if you just look at the history, right, through all the topical reports, all the changes that were made. But when you look at it through a risk lens, it may not make that big a difference. And that's why the tool allows that flexibility to step back and say, why are we managing having oversight of all of these little changes when at the CDF level there may not be a big -- you may not see it.

MEMBER PALMTAG: Well, let's use an example of critical power ratio or something like that. We have all kinds of programs. We all these correlations. We have all this verification and validation.

A lot of time is put into CPR validation, verification. But in the end, if you do change your CPR slightly, it's still not going to change your CDF. So it just seems to invalidate all that work we've done and the CPR correlations and verification validation.

If you look at it and say, well, this doesn't change a CDF. Therefore, it's not meaningful.

To me, it seems like a lot of details in V&V is kind of being swept away if you just say, well, it's not going to change your CDF. Maybe that's the intention. I don't know.

VICE CHAIR PETTI: Well, I tend to look --

MEMBER PALMTAG: I'm just trying to understand this.

VICE CHAIR PETTI: I tend to look at it as the level of NRC oversight. I still think there's value in validating these things because it's good engineering practice. But the question is, do you have to say mother may I for every single change when it may not have the big effect? I still think you do just because --

(Simultaneous speaking.)

MEMBER PALMTAG: Like I said, a lot of details are going to be lost if the only thing you're looking at is the CDF. So --

MEMBER HARRINGTON: This is Craig. Again, Scott, are you suggesting that tying it to change in CDF maybe is too cross of a criteria?

MEMBER PALMTAG: Exactly, yeah.

MEMBER HARRINGTON: Not to be tied to

something lower?

MEMBER PALMTAG: Example, CPR. You can make a lot of changes to CPR that's not going to affect your -- it's not going to make it all the way up to change a CDF. So I'm just kind of worried. A lot of change is going to get pushed under the rug once you -- if that's the only thing you're looking at is the CDF. I'll leave my comment at that.

MR. HELLER: Well, I think it's a good comment. But I would point out that the VVUQ program within the proposed Section 221 is not necessarily directly tied to seeing an impact on CDF.

MEMBER ROBERTS: This is Tom. That's exactly what I would ask. Reading through the FRN, it honestly tied between the VVUQ part of the this 50.59 change and CDF. There's two different changes being made. One is any change that you want to propose via 50.59 to risk. But the second part, VVUQ, seems to be risk-independent.

It does use the term risk-informed one place that I found that defined what's risk-informed about the VVUQ process other than depending on the consequence of the magnitude of the change. You need the VVUQ process reviewed by different entities. But

it doesn't seem to tie to CDF at all. It seems to tie to a somewhat deterministic evaluation of how big a deal an error in that code might be. Is that right?

MR. HELLER: Mm-hmm. Yes, I should say yes. Yes.

MR. MOULTON: Yes. These two changes were tied together because they affect the same section of Part 50. That's really the only reason.

VICE CHAIR PETTI: Any other questions?

MEMBER HARRINGTON: Nothing here in the room, Dave.

VICE CHAIR PETTI: Yeah, let's keep moving. We've got 30 minutes before we're scheduled for a break.

MR. MOULTON: I want to thank you again for the opportunity to present.

MR. DOYLE: Thank you, Kevin and Charles. My name is Daniel Doyle. I am the rulemaking project manager. For this activity, I have a few more slides to conclude the staff's presentation this morning.

So I wanted to talk about the next steps. So the NRC last week publicly released a preliminary version of the proposed rule, Federal Register notice that is the unsigned, unofficial version, but intended

to provide stakeholders an early review of that document. There was an NRC press release.

So the actual proposed rule will be published in the Federal Register soon. The estimated publication date is next week. There will be a 45 calendar day comment period, and then we'll go through the typical rulemaking steps for a final rule, evaluating public comments, preparing the updated and supporting documents that includes a congressional review determination and review under Executive Order 12866 as we did for the proposed rule.

The final rule is then scheduled for publication near the end of this calendar year and would have a 30-day effective date. The contents of the proposed rule package, so again, just to be clear, this is not publicly available right now. We have released the preliminary version.

The documents that will all be publicly available by the time the proposed rule is published in the Federal Register include the documents shown on the slide here. So there's the Federal Register notice obviously, regulatory analysis evaluating cost and benefits, environmental assessment, and then significant number of supporting documents, some that

were referenced in the topics discussed today. So there are 22 draft regulatory guides, one interim staff guide, interim staff guide update, one NUREG, a standard review plan, SRP update. And then for Paperwork Reduction Act requirements, we have a supporting statement and burden tables. We'll also be releasing an unofficial red line version showing how the proposed rule would make changes to the Code of Federal Regulations. And that concludes the staff's presentation.

VICE CHAIR PETTI: Any other comments, members, questions?

MEMBER HARRINGTON: I see none here, Dave.

VICE CHAIR PETTI: Okay. At this point then, we'll go ahead of schedule. But I think that's good. I think we could go to public -- we have two options. We could go to public comment or we could do our break early. What does those in the room feel?

MEMBER HARRINGTON: Everybody okay doing public comments now? I see heads nodding, Dave.

VICE CHAIR PETTI: Okay. Let's do public comment. Again, raise your hand on Teams or if you're in the room. Line up the questions.

MR. BURKHART: Yeah, Dave. This is Larry

Burkhart, if I may.

VICE CHAIR PETTI: Sure.

MR. BURKHART: If you are online, some folks have already raised hands, please continue to do so. We will take your comments in sequence. Okay. With that, Dave, I'll turn it over to you. We have a few folks that have raised their hands.

VICE CHAIR PETTI: Yeah, I see Jon, JF. I can't see the name, the first one. Go ahead.

MR. FACEMIRE: Jon Facemire with the Nuclear Energy Institute.

First, I do want to applaud the staff for this really monumental effort. There's a ton of good stuff in here. The question that I had was on the determinate thresholds. I think the thresholds chosen are reasonable and align with a lot of precedent. But I wanted to talk and ask about the coordination with other ongoing efforts. So the licensing modernization project, their definition of beyond design basis event goes down to 5e to the minus-7.

So if this proposed rule were to go through as is, would they be able to use it to justify using 1e to the minus-6 as a cutoff frequency instead of the 5e to the minus-7? And then similarly, Part 57 has

a Table 12-1 in the NUREG 2271 guidance that has quite different thresholds for what needs to be considered in the design basis. So the specific question is about LMP and whether they could use this to justify the 1e to the minus-6 threshold. But the broader question is, how is NRC coordinating across the various rulemaking and licensing pathways that are available?

VICE CHAIR PETTI: Thank you.

MEMBER HARRINGTON: This is Craig Harrington. Just to point out it's not a question and answer session. So if you've got questions like that, the proper forum for those is the public comment period in the FRN that's being released soon, I believe. The next hand --

MR. DOYLE: I'm sorry. This is Dan Doyle. If I could just provide another piece of information. Probably should've mentioned in my previous slide. The NRC staff will be having a public comment -- I'm sorry, will be having public meetings to discuss the proposed rule, and there will be some opportunity for clarification questions.

Comments should still be submitted via regulations.gov. But I just wanted to point out that will be having a public meeting. Notices will be

available on the public website. So just want to keep that in mind.

MEMBER HARRINGTON: Yes, thank you. Go ahead, Dave.

VICE CHAIR PETTI: Yeah, I think I saw Ed Lyman next.

MR. LYMAN: Thank you. This is Edwin Lyman from the Union of Concerned Scientists. We really haven't had sufficient opportunity to review the implications of this extensive rule. But in terms of what's been discussed this morning, again, I would like to point out the public confidence is still important component of what the NRC is responsible for.

And, to my mind, public confidence depends on having a set of uniform common standards that do not depend on an enormous amount of staff discretion and subjective judgment that these were based on, again, a lot of thought and experience in developing those common standards and hearing referring to things like the GDCs, the Appendix B programs. Those are aspects where they're deterministic but important that there is some deterministic core to the basis for nuclear safety and what the NRC is doing.

And to go too far in the direction of

relying on PRA going beyond what the PRA policy statement judge was acceptable, I feel like what's happening now is the staff is taking this opportunity to kind of steamroll over all the objections and concerns that have been raised -- that have delayed or raised questions about implementation of some of these more drastic applications of PRA beyond which -- where the applications go beyond the state of the art of the technology.

And so by moving away from these common standards, clear, deterministic, basic safety standards for nuclear power plants, I think this is going in the direction of chaos. And certainly as a member of the public, I'm losing confidence in the ability to understand what that common safety basis is in terms of the NRC carrying out its statutory requirements.

So, again, I'm deeply troubled by these exceptions by using graded approaches in PRA to water down or move away from that common set of deterministic standards that I think shows that there is at least a uniform level of safety across the fleet and for new plants. So those are my initial comments. Thank you.

VICE CHAIR PETTI: Niko?

MR. McMURRAY: Awesome, thank you. Can you hear me?

VICE CHAIR PETTI: Yeah.

MR. McMURRAY: Great. This is Niko McMurray with ClearPath. More of an administrative question. I know during the ACRS meeting on Part 57, the staff and the ACRS accelerated the publication of the meeting transcripts just due to the short timelines of the rule -- of the public comment period for the rule. So I'm wondering if the staff will try to also accelerate that timeline to publish the meeting transcript to support stakeholders providing feedback during the public comment period.

MEMBER HARRINGTON: This is Craig Harrington. I see a thumbs up over here from our ACRS staff. So yes.

MR. McMURRAY: Awesome. Thank you much. I appreciate that. Thank you.

VICE CHAIR PETTI: I don't see anymore hands online. Are there people in the room that have a comment?

MEMBER HARRINGTON: There are no people in the room lining up or raising their hand to make comments.

VICE CHAIR PETTI: Okay. Well, with that, then I think it's time for our break. Let me go back to look at the agenda. We're going to public comment and then closing remarks by the chairman. Is that you?

MR. BURKHART: This is Larry Burkhart. Dave, that's you, the chairman of the topic. And then, of course, you can read the letter in when you -- well, deliberation and then read the letter in when you're ready.

VICE CHAIR PETTI: Right, right. Okay. So let's take a 15-minute break. We'll come back at 10:45 and talk about what we've heard. And I do have a draft letter to read in. Thank you.

(Whereupon, the above-entitled matter went off the record at 10:28 a.m. and resumed at 10:45 a.m.)

VICE CHAIR PETTI: Okay. So, members, I guess we should just open up the floor to any broad concerns or points that you want to make. I will say that I do have a draft letter and I thank many of the members who took the time to read it. It's a long draft letter given the depth and breadth of the scope that we've heard about this morning. Does anybody have any

questions or concerns, comments?

MEMBER SUNSERI: This is Matt. I don't think we have any. I think we've all looked at the draft proposal there and it seems like that'd be a good way to start the deliberation.

VICE CHAIR PETTI: Okay. So, Sandra, can you put the draft letter up? And, Larry, do we have the court reporter but this in the record when I read it in?

MR. BURKHART: Yes. Yes, Dave. This is Larry. You should read it in, and then we'll let the court reporter go.

VICE CHAIR PETTI: Okay, right. Okay, good.

Okay. Subject: Proposed Rulemaking on Modernizing Reactor Licensing, Safety Oversight, and Siting Practices. Dear Chairman Nieh, during its 737th meeting held from July 8th through 10th, 2026, the Advisory Committee on Reactor Safeguards discussed with the NRC staff this draft rulemaking which is part of the agency's broader effort to conduct a review and also a vision of its regulations to modernize its regulatory framework. We also benefitted from a number of closed engagements with the staff on the

specific topics addressed in this letter over the past six months and from the referenced documents.

Conclusions and recommendations. One, the proposed regulatory changes implement a graded approach that reflects the differing risk profiles of advanced non-light water reactors, small modular reactors, and microreactors relative to large light water reactors.

Two, these changes increase flexibility for applicants by removing prescriptive and in some cases outdated requirements and by emphasizing risk-informed technology inclusive and performance-based approaches.

Three, the rule continues to acquire adequate defense-in-depth and sufficient safety margins, two key cornerstones of nuclear safety. For advanced reactors, however, these attributes should be evaluated in the context of the overall design rather than solely against precedents established for the existing LWR fleet.

Four, because of the breadth of the rulemaking changes and the speed of development and implementation, the potential exists for contradictions or inconsistencies with the changes

that were made. During the public comment period, the staff should use the time to look for adverse interactions, gaps or contradictions, and correct them before the rule is finalized.

Five, in implementing these changes, it will be important to ensure that the drive to eliminate prescriptive requirements does not inadvertently introduce new vulnerabilities. While we encourage justified reduction and the excess of conservatism that has historically been associated with overly prescriptive approaches, there are also pitfalls to be avoided in the other direction. Precise navigation will be required to avoid the perils of either side.

And, six, we support the rulemaking at a high level and offer detailed comments for your consideration in the body of this letter.

Background. On May 23rd, 2025, the President signed Executive Order 14300 ordering the reform of the Nuclear Regulatory Commission. Section 5 of the EO directs the NRC to conduct a comprehensive review and also a revision of its regulations in Title 10 of the Code of Federal Regulations and associated guidance documents.

This proposed rulemaking fulfills that

directive by streamlining and clarifying requirements for licensing, design certification, emergency preparedness, quality assurance, site criteria, fuel enrichment, and other areas. The overarching purpose is to enhance flexibility and efficiency, reduce unnecessary regulatory burden, and enable risk-informed performance-based alternatives while maintaining safety and security. This proposed rulemaking affects 10 CFR Parts 2, 50, 51, 52, 53, 54, 71, and 100.

The Committee focused its review on the unique, novel, and noteworthy aspects of the following five key topics from the proposed rulemaking to modernize NRC's regulatory framework following the ACRS review flow chat in the ACRS guidance and best practices, enhancing the flexibility of reactor site criteria, determinant and data-backed thresholds for reactor safety assessments, alternative risk-informed and performance-based acceptance criteria for 10 CFR Part 50 and 52, quality assurance, 10 CFR Part 50, Appendix T, and risk-informing, 10 CFR 50.59. This doesn't look like the right version, Sandra. I struck Line 63.

The Committee did not re-review topics

integrated in the proposed rule that were previously provided to the Commission on -- such as the February 22nd, 2025 letter report on the proposed rulemaking on increased enrichment of conventional and accident tolerant fuel designs for light water reactors. In addition, the Committee did not review changes to operate a training in large part because the safety significant changes were very similar to those in Part 53 that we commented on previously. Enhancing the flexibility of reactor site criteria, the smallest physical footprint reduced radiological risk and enhanced safety features of advanced reactors, including small modular reactors and microreactors relative to large LWRs supported graded approach through site hazard characterization.

This gradation would increase flexibility and reactor siting criteria in 10 CFR Part 100 and enable deployment of these reactors across a broader range of locations in the United States. The proposed rule establishes a tiered framework for reactor siting. Since 1997, 10 CFR Part 100, Subpart A has applied to test reactors while Subpart B has applied to all power reactors.

Under the proposed rule, Subpart A would

be repurposed as an intermediate tier, bridging the proposed Part 57 framework for microreactors and Subpart B for larger reactors. An unmitigated dose consequence metric is used to characterize the hazard of these facilities and establish tiers for the different reactors. A Tier 1 reactor is a power reactor having an unmitigated dose consequence less than 25 rem TEDE at the site exclusion area boundary.

A Tier 2 reactor is a power reactor having an unmitigated dose consequence greater than 25 rem TEDE at the site EAB. Draft Regulatory Guide DG 4036 establishes a graded approach to site characterization with a goal to achieve acceptable and balanced risk by applying increased design requirements based on increasing consequence of failure. Thus, system structures -- structures, systems, and components with higher consequences of failure need to be designed to withstand lower frequency events.

While some of the pieces for a graded approach currently exist in Reg Guides, no comprehensive NRC approach exists. In establishing this graded approach, the draft guidance drew upon current guidance on site characterization and external hazards used for LWRs non-power production and

utilization facilities, NPUFs, and research and test reactors.

Yeah, hold on. Okay. Where are we here? Yeah, Line 94.

The DG also provides guidance on performing the unmitigated consequence analysis with an appendix describing relevant considerations necessary to calculate airborne radioactive releases. The DG establishes nuclear siting categories based on the numerical values of the unmitigated release consistent with guidance in ANSE and ANS 2.26-2004. These different categories provide gradation beyond the reactor tier and are tied to associated seismic design categories.

In addition, the DG establishes three levels of graded site investigations, comprehensive, limited, and minimum site characterization the areas of geotechnical engineering, geology, and hydrology. Other draft Regulatory Guides on site specific ground motion, meteorological monitoring, and flooding have been updated to incorporate graded approaches consistent with the overall siting approach.

ACRS comments. One, the graded approach to siting and site characterization is a worthy

objective as it fundamentally attempts to account for the different hazards/risk profiles associated with advanced reactors, small modular reactors, and microreactors compared to large LWRs.

Two, the graded approaches to geology, hydrology, geotechnical engineering, and other site characteristics are very reasonable. The new guidance on use of offsite meteorological data and the updates to flooding guidance are also reasonable and appropriate. The material at risk approach has worked well for the Department of Energy non-reactor facilities but may not work as well for the range of commercial reactor technologies and sizes without additional guidance and what constitutes an unmitigated event.

The draft guidance, DG 4036, currently knows that widespread failure of SSCs is to be assumed based on a severe external hazard. And only inherent chemical and physical characteristics of the material at risk can be considered. Maximum temperature due to loss of cooling can also be considered in the evaluation. And although the guidance is less clear related to reactivity events that can be very important in some microreactor designs, for example, a reactivity

addition unmitigated by scram could lead to an energetic event with consequences significantly more severe than those caused by loss of cooling. Such anticipated transient without scram type events are required to be addressed per 10 CFR Part 50. And it's not clear how or whether they ought to be addressed for the new 10 CFR Part 100 requirements.

We suggest that the guidance be clarified to explain whether and how reactivity transients without scram, severe reacting excursions ought to be assessed as part of an unmitigated dose consequence evaluation.

Four, the actual words used to describe the unmitigated dose value, the careful consideration from a public perspective, it is merely a metric calculated under very artificial conditions with the goal of establishing a level of robustness of SSCs necessary to reduce the hazards of the facility to a safe level. However, the perception by the public may be quite different when seeing such large numbers compared to the common perception about the safety of microreactors advanced non-LWRs and SMRs.

Five, alignment of the nuclear safety categories to seismic design categories, SDCs, allows

optimization of the seismic design relative to the hazard at the beginning of the design process well in advance of any detailed seismic analysis or seismic probabilistic risk assessment being performed. This gradation is appropriate given the impact of seismic requirements on the complexity of the facility design. The use of, quote, nesting, lower SDC SSCs inside a number of more robust higher SDC SSCs as described in the Reg Guide will also help in overall seismic design optimization. The approach is appropriate. It's worth considering the impact of this optimization on other external hazards such as tornado and high wind missiles.

MEMBER KIRCHNER: Dave, this is Walt Kirchner. Could we just take one pause on reading the letter real quick and then --

(Simultaneous speaking.)

VICE CHAIR PETTI: Okay, okay.

(Pause.)

(Off-the-record comments.)

VICE CHAIR PETTI: Okay. Determinate and data-backed thresholds for reactor safety assessments. The purpose for using determinate and data-backed thresholds in reactor safety assessments is to ensure

that safety assessments focus on credible and realistic risks. Establishing clear thresholds for initiating events across a spectrum of event frequencies together with qualitative screening criteria is expected to provide a sound framework for categorizing initiating events, identifying design basis events, and applying a graded approach to beyond design basis events. Establishing what are credible realistic risks also involves consideration of operating experience, PRA insights, and guidance from sources, including the increased enrichment rulemaking, PRA insights, guidance from sources including the increased enrichment rulemaking, the licensing modernization project approaches, non-LWR PRA standard, international guidance, and practices as well as previous NRC precedents.

Draft Guide DG 1454 provides such numerical thresholds as well as qualitative screening criteria. Specifically, design basis events are defined as those caused by initiating event frequencies of occurrence that are greater than 10 to the -4 per reactor year based on the upper 95 percent confidence bound by frequency and certainty and include AOOs. For these events, the traditional safety analyses in

Chapter 15 use conservative methods to demonstrate the capability of safety-related structure, systems, and components during design basis events.

The category also considers the effects of natural phenomena. Beyond design basis events, BDBEs, are those on the threshold of credibility. BDBEs are caused by initiating events with frequencies of occurrence between 10 to the minus-4 and 10 to the minus-6 per reactor year based on the upper 95 percent confidence bound on frequency uncertainty. All are required by regulation to be considered in the licensing basis, for example, ATWS station blackout and large grade loss of coolant accident events.

Safety assessments in this range want some realistic conditions and use best estimate methods along with evaluation of key uncertainties to take credit for non-safety related SSCs in assessment of plant response if they are not affected by the initiating event similar to the LMP. Three, demonstrate adequacy of defense-in-depth and address cliff edge effects. And four, justify that safety margins are sufficient to support a reasonable assurance of safety finding.

Non-credible events are those with

initiating event frequencies, 10 to the minus-6 per reactor year and are not required to be evaluated by this proposed regulation. ACRS comments, we consider this overall approach to be viable enough of the following for consideration. One, inconsistent terminology, the new definition of design basis event along with the existing term, design basis accident, and 10 CFR Part 50 is not consistent with other regulatory uses of these terms. For example, the LMP defines and uses these two terms in a different way than the new 10 CFR Part 50 definition.

Additionally, the term DBA would remain undefined in 10 CFR Part 50, making them clear what distinction is being drawn for instances with DBA versus DBE is used within 10 CFR Part 50. We suggest the staff consider actions that may be warranted such as additional guidance to avoid confusion in implementing the new definitions. Two, adequacy of defense-in-depth, the draft Reg Guide, Section 3.8 states that compliance with 10 CFR Part 50 or 52 inherently provides adequate defense-in-depth.

This is true for the large LWRs for which 10 CFR Part 50 was originally written. However, it is not evident how this inherently remains true for

advanced reactors that use 10 CFR Part 50 or 52. For example, while Part 50 includes specific beyond design basis requirements such as station blackout and ATWS which provide meaningful regulatory guardrails, those guardrails were constructed from LWR operational surprises and reflect vulnerabilities of that specific technology.

For advanced reactors which large epistemic uncertainties characterized by novel coolant chemistries, passive safety systems, and integrated architectures, evaluating these events may not be necessary but not obviously sufficient. For this reason, non-LWRs using LMP require a separate assessment of defense-in-depth even though such LMP-based applications generally use 10 CFR Part 50 or 52. Advanced reactor applicants who choose to use 10 CFR Part 50 or 52 without LMP would appear to also need a separate assessment of defense-in-depth.

We suggest the staff clarify the conditions under which 10 CFR Part 50 or 52 compliance inherently assures adequacy of defense-in-depth and add guidance for defense-in-depth assessments for plant designs that do not meet these conditions. Three, assumption of independent failures, the draft

Reg Guide, Section 3.1 provides significant limits on when an independent failure of a non-safety system can be included within an initiating event. It is not clear why any restrictions would be placed on an applicant choosing to assume additional failures.

This is generally conservative. And why the regulator would put restrictions on such cases is unclear. We suggest this section be clarified to better explain the intent of the restrictions.

Quality assurance, quality assurance requirements in Appendix B of 10 CFR Part 50 were established in 1970 to prevent, design, fabrication, and construction of nuclear power plants. Since that time, a number of limitations of Appendix B have been identified, including it is not risk-informed and performance-based, it is not able to tailor the requirements for low risk activities. It uses updated language and has no explicit direction for digital systems out of manufacturing or software QA.

Many vendors are unfamiliar with Appendix B, and it is not aligned with international standards. Thus, there is a need for a modernized risk-informed QA framework that retains the safety rigor of Appendix B while allowing graded application. This new

optional framework would be a new Appendix T of 10 CFR Part 50.

Appendix T provides performance-based QA criteria of explicit direction on the use of a graded approach. This new appendix is streamlined relative to Appendix B and its use is voluntary. It is intended to provide a technology inclusive graded approach for implementation. There are three entrance conditions to use Appendix T.

It must be an nth-of-a-kind plant. There must be no change in safety SSCs compared to the first-of-a-kind plant. And the program must include procedures and work processes for implementation of Appendix T.

Appendix T defines requirements for a quality management system instead of the quality assurance program description used in the past. It includes QA requirements for software and may eliminate the need for NRC oversight of suppliers and vendors. The requirements are expressed as 11 separate criteria grouped around three topics, management, performance, and assessment.

The requirements align with Appendix B and are more consistent with international quality

standards. However, any gaps between industry standards and the QA requirements must be addressed by the applicant. Two questions are included in the FRN on restriction of Appendix T to nth-of-a-kind plants and on the changed process for the QMS under 10 CFR 50.54(a)(5).

ACRS comments. Quality assurance at nuclear power plants and fuel reprocessing facilities creates an underpinning foundation for nuclear safety by ensuring that planned and systematic actions necessary to design, construct, and operate these facilities, provide confidence that structures, systems, and components will perform satisfactorily and reliably in service. The proposed Appendix T for NOAK facilities adequately fulfills the subjective in lieu of 10 CFR Part 50, Appendix B and offers adequate defense-in-depth to nuclear safety. Because of the broad integration of quality assurance with safe operation of these facilities, it is important to have consistency and clarity in quality assurance requirements.

We note the following comments as opportunities to improve the consistence and clarity of requirements in the proposed Appendix T. One,

Appendix T has three requirements, one of which is that the facility adopting Appendix T is nth-of-a-kind with sufficient fidelity to the first-of-a-kind baseline facility. Staff discussed that Appendix T has been mapped with and fully supports all attributes of quality described in Appendix B.

There are additional benefits in Appendix T such as guidance on software quality assurance. Responses to one of the FRN questions may provide an opportunity to reconsider this limitation and allow Appendix T as a viable option for first-of-a-kind applicants. This approach would eliminate potential inconsistencies between operators of a specific design and suppliers that may be required to have multiple quality assurance programs to support first-of-a-kind and nth-of-a-kind applications.

Two, the applicability of the draft Appendix T to SSCs that are designed as non-safety related but safety-significant, NSRSS, is inconsistent throughout the draft and may warrant clarification. For example, its applicability to SSCs deemed to be, quote, non-safety related with special treatment, NSRST, per Reg Guide 1.233 endorsed LMP, are those categorized under regulatory treatment of non-safety

systems, RTNSS, for passive advanced LWRs for SRP 19.3 guidance is unclear, even though these SSCs are selected using similar criteria to those stated for NSRSS.

Additionally, the software QA requirement for digital control systems is limited to safety functions, a term that is not defined in Appendix T, but is defined in IEEE 603-1991, endorsed by 10 CFR 50.55a(h). As features, quote, essential to maintaining plant parameters with an acceptable limits established for a design basis event, quote.

Limiting the digital I&C requirement in this manner means the requirement will not apply to NSRSS-type digital I&C SSCs because such SSCs are, quote, relied on to achieve adequate defense-in-depth to perform risk-significant functions, not to directly mitigate design basis events.

On a broader scale, there's an opportunity for harmonization across the nuclear industry regarding multiple approaches to quality assurance found in 10 CFR Part 50, Appendix B, proposed Appendix T, nuclear quality assurance certification NQA-1, DOE order on quality assurance 14.1, or DOE 10 CFR 830 Subpart A, quality assurance requirements, quality

management system requirements, ISO 9120.15, and other lesser used quality assurance standards. A harmonized standard for quality assurance requirements from the regulatory for nuclear power plants and fuel reprocessing facilities and their suppliers will enhance consistency and clarity across the sector, thus strengthening the foundation for nuclear safety.

Alternative risk-informed and performance-based acceptance criteria. Many of the regulations include both numerical and non-numerical limits as acceptance criteria that restrict the ability of applicants to propose approaches without an exemption. The rule would allow expanding use of risk-informed and performance-based criteria, enabling the focus to be on safety outcomes and risk significance instead of rigid rules.

It would also allow deviations from general design criteria with justification in the application instead of using the exemption process. The rule still requires the design to maintain adequate defense-in-depth and sufficient safety margins. The applicant is urged to vet such proposals to pre-application engagement if they choose to use these new flexibilities.

Associated new draft guidance has been developed by the staff, DG 1464. The guidance notes that the applicant should discuss what conditions will be addressed with the criteria and specify the proposed alternative criteria, including quantitative values of performance targets. Any sufficient or adequacy evaluations need to be developed in the context of an integrated decision making panel using guidance in Reg Guide 1.201. The draft guide also discusses the need to consider the cumulative impact of the proposed alternative criteria.

ACRS comments. One, the rule language requires maintenance of adequate defense-in-depth and sufficient safety margins, key cornerstones of nuclear safety. This is important. However, we observe that advanced non-LWR small modular reactors and microreactors present different risk profiles than large LWRs.

Two, we support the approaches described in the draft Reg Guide having a strong sound technical basis for alternative criteria substantiated by the different risk profiles of an advanced non-light water reactor, small modular reactor, or microreactor relative to large light water reactors should enable

efficient review and approval. This confirming 10 CFR 50.59.

The changes to the rule will allow the use of quantitative risk metrics such as core damage frequency, CDF, and large early release frequency, LERF, and the changes, delta CDF and delta LERF, when performing assessments. Early implementation of 10 CFR 50.59 did not use such quantitative criteria in their assessments because of the lack of experience with risk assessments. Today's risk assessments are more mature.

Thus, such quantitative assessments are consistent with decades of evolving practice on risk-informed regulation at NRC. As part of the process that changes must not result in large changes to the risk metrics and must maintain adequate defense-in-depth and sufficient safety margins. The delta CDF must be less than 5 times 10 to the minus-7 per reactor year and the delta LERF must be less than 5 times 10 to the minus-8. And the cumulative risk metric must be less than 10 to the minus-4 for CDF and less that 10 to the minus-5 for LERF.

The staff will consider other metrics for non-LWRs. The use of the PRA directly in these

evaluations raises issues related to the quality of the PRA, the associated uncertainty assessment, and the impact of any updates to the PRA from the approved licensing baseline.

However, decades of experience from occurring peer reviews on site specific PRAs of the current operating fleet demonstrate that such concerns can be mitigated. The rule also allows for changes to models used in evaluations to no longer require NRC review and approval if the applicant hasn't approved verification, validation, and uncertainty quantification, VVUQ program. Such approaches are developing in other parts of the government, for example, the Food and Drug Agency and the Federal Aviation Administration, and represent an emerging consensus type of standard for computational modeling.

This change in regulation results in a shift away from evaluating each individual model directly into the evaluation of the process by which models are determined to be credible, trusted through a structured VVUQ program. It allows applicants to evolve their models to incorporate advanced modeling methods and updates to operating platforms. The staff has developed a draft Regulatory Guide, DG 1468, that

describes the high level attributes of such an approach.

The program is centered around establishing risk levels for model and simulation activities. The rigor of the VVUQ assessment should reflect the importance of the model's role in decision making and the risk associated with errors in the results. ACRS comments, one, the approach expands the options available to applicants to make changes and plant design and operation to incorporate quantitative risk results and their assessments.

Such changes require a risk assessment supplemented by an evaluation of adequacy of safety margins and defense-in-depth. While the regulation and guidance changes provide significant detail on the PRA quality and allowable frequency levels to use for the risk-informed decision, very little guidance is provided to define evaluation criteria the licensee is expected to use in determining whether a risk-informed change has maintained acceptable safety margins and defense-in-depth. For example, the draft Regulatory Guide calls out the following: documentation and justification of how defense-in-depth is maintained and documentation and

justification of how safety margins are maintained.

Determining whether these criteria are met requires significant qualitative judgment which is currently validated by the NRC staff in their review of risk-informed license amendment requests submitted by Reg Guide 1.174. We recommend that the staff focus on adequacy and the safety margin in defense-in-depth assessments in their early regulatory audits of inspections of licensee self approvals, granted using this new provision and plan to adjust the guidance as warranted based on these audits.

Two, for the parts of the rule associated with computational modeling, the lack of detailed review of a model change and its results is counterbalanced by the need for an approved VVUQ program. While legacy vendors would probably be attracted to such an approach given their significant operating experience and test data, this may be difficult for advanced reactors to consider given their lack of operating experience. The guidance has largely focused on the operating fleet. The guidance may need to be updated and broadened to consider evaluation models for advanced reactors in the context of the limited operating experience and validation

technical bases.

Three, the use of, quote, modeling and simulation, unquote, in the draft guide is purposely quite broad. As sentence or footnote in the guide that would explicitly define the term as applying to a range of computational work for the reactor, including, for example, traditional safety evaluation models, PRAs, digital twins, and even structural design and component performance analyses would improve the clarity. Summary, the proposed regulatory changes implement a graded approach that reflects the risk profiles of advanced non-light water reactors, small modular reactors, and microreactors relative to large light water reactors.

These changes increase flexibility for applicants by removing prescriptive and in some cases outdated requirements and by emphasizing risk informed, technology inclusive, and performance-based approaches. The rule continues to require adequate defense-in-depth and safety margins, two key cornerstones to nuclear safety. For advanced reactors, however, these attributes should be evaluated in the context of the overall design rather than solely against precedence established for the

existing LWR fleet.

Because of the breadth of the rulemaking changes and the speed of development and implementation, the potential exists for contradictions or inconsistencies with the changes that were made. During the public comment period, the staff should assure that any such inconsistencies are identified and rectified before the fuel is finalized. We support the rulemaking at a high level and offer detailed comments for consideration in the body of this letter.

As the NRC advances this modernization effort, we are encouraged by the meaningful progress reflected in this proposed rulemaking. We congratulation the staff who developed these changes in a short amount of time. Further, we observe that the senior staff is supportive and engaged.

After the Battle at El Alamein in World War II, Winston Churchill said that it was not the end or even the beginning of the end, but that perhaps it was the end of the beginning. We have the same cautious hope. We will remain interested in this activity and would like to be kept aware of the progress. We are not requesting a response to this letter. Sincerely,

Greg Halnon.

CHAIR HALNON: Hey, Dave, take a breath.

VICE CHAIR PETTI: Yeah.

CHAIR HALNON: So, it is just coming up to 11:30, 11:22 right now. And the approach that I would like to take on this letter is to take topic by topic to comment on. Normally, we go to the recommendations and conclusions first. But I think that's going to fall out of the topic by topic because the more detailed comments are in the topics.

We will first work on the aspect of the overall high-level comments in a topic, talk through the comments and recommendations. And we're not going to worry too much about the wording at this point.

We'll do the wording later. But I want to make sure that we're aligned on the comments and any high level issues that you have with it. So before we move on, is there any comment on that? I mean, I'm open to -- it's such a long letter. Doing it line by line at this point would be futile. Okay.

So I think, Shakil, the court reporter, I think you're released for the rest of the day. We need you back at 10:30 a.m. tomorrow for a continuation of another topic. Shakil, did you get that?

COURT REPORTER: I did.

CHAIR HALNON: Okay. We'll see you back at 10:30 tomorrow. If you have any questions, get a hold of Larry or Weidong. Thank you for your time today.

(Whereupon, the above-entitled matter went off the record at 11:24 a.m.)

Modernizing Reactor Licensing, Safety Oversight, and Siting Practices Proposed Rule

Advisory Committee on Reactor Safeguards Full Committee Meeting

RIN 3150-AL44; NRC-2025-0975

July 8, 2026

Purpose

Provide an update on topics of interest from this proposed rule for the Advisory Committee on Reactor Safeguards Full Committee

Background

- EO 14300, “Ordering the Reform of the Nuclear Regulatory Commission,” Section 5, “Reforming and Modernizing the NRC’s Regulations,” directs the NRC to undertake a review and wholesale revision of its regulations and guidance documents.
- This rulemaking addresses Sections 5(f), 5(h), and 5(i) of EO 14300 and other items from the wholesale revision that the NRC determined would enable adding additional power generation to the electrical grid.
  - Section 5(f): “[e]stablish stringent thresholds for circumstances in which the NRC may demand changes to reactor design once construction is underway”
  - Section 5(h): “[a]dopt revised and, where feasible, determinate and data-backed thresholds to ensure that reactor safety assessments focus on credible, realistic risks”
  - Section 5(i): “[r]econsider the regulations governing the time period for which a renewed license remains effective, and extend that period as appropriate based on available technological and safety data”

Purpose of Proposed Rule

- Modernize NRC regulations to improve efficiency, clarity, and flexibility in reactor licensing, safety oversight, and siting practices.
- Implement EO 14300 sections 5(f), 5(h), and 5(i), and the ADVANCE Act
- Propose risk-informed, performance-based approaches to enable innovation for advanced nuclear technologies.
- Reduce unnecessary regulatory burden while maintaining reasonable assurance of adequate protection of public health and safety and common defense and security.

Topics

1. Incorporation of Streamlined Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants (Appendix T)
2. Enhancing Flexibility of Reactor Site Criteria
3. Determinate and Data-Backed Thresholds for Reactor Safety Assessments
4. Alternative Risk-Informed and Performance-Based Acceptance Criteria for All Part 50 and 52 Regulations
5. Risk-Informing 10 CFR 50.59 and Allowing Flexibility for Changes to Methods

Quality Assurance – 10 CFR Part 50 Appendix T

Overview of Proposed Appendix T

- As a result of concerns raised, the NRC is proposing to add Appendix T to 10 CFR Part 50 as a voluntary alternative to Appendix B to 10 CFR Part 50.
- The proposed Appendix T would draw on international quality assurance (QA) standards to incorporate the following elements:
  - Performance-based QA criteria that provide explicit direction on use of a graded approach for applying QA requirements to structures, systems, and components (SSCs) relative to their safety and risk contributions to the overall nuclear facility.
  - QA requirements specific to software used in digital items and for design and analysis.
  - QA terminology and methodologies used across various safety critical industries and in international standards for quality management thus allowing applicants to leverage cross-industry and global supply chains.
  - Provide for increased licensing certainty by removing the need for NRC oversight of suppliers and vendors of products and services related to nuclear power plant and fuel reprocessing plant SSCs subject to this proposed rule.

Eligibility Criteria for Appendix T

To reference this appendix in an application for a construction permit, operating license, or combined license, an applicant must demonstrate it meets the following criteria:

- The application is for an nth-of-a-kind construction permit, operating license, and combined license that identifies the first-of-a-kind reference plant in the application and any departures from the first-of-a-kind reference plant;
- Departures from the first-of-a-kind reference plant do not change the classification, design, and method of manufacture, construction, and operation of SSCs within the scope of this appendix; and
- Procedures and work processes for implementing the requirements in this appendix have been established and included in the application.

General Requirements – Integrated Quality Assurance Program

- Establish QA Program, document QA program in the Quality Management System (QMS), and submit QMS to NRC for review and approval
- QMS must implement QA requirements in Section IV of Appendix T for
  - Management
  - Performance
  - Assessment
  - Software used in design and analysis and digital items important to safety
- Implementation of approved Quality Management System
- QA Program must cover safety-related SSCs and nonsafety-related safety significant SSCs
- Implement a graded approach

Questions in FRN

- Eligibility Criteria: How can the NRC expand the applicability of Appendix T to first-of-a-kind plants and existing licensees, including any guardrails or restrictions that may be desirable (e.g., vendor inspections for first-of-a-kind and new suppliers for nth-of-a-kind)? Should the NRC consider alternative approaches to Appendix T such as endorsement of ISO 19443 or changes to Appendix B?
- QMS Change Criteria: Should the change process have a graded approach or mirror 10 CFR 50.54(a)(4) in which only changes that reduce the commitments be submitted to the NRC for prior approval?

Enhancing Flexibility of Reactor Site Criteria

Enhancing Flexibility of Reactor Site Criteria

- Proposed rule would repurpose Part 100 Subpart A as a middle tier between Part 57 and Part 100 Subpart B requirements to increase regulatory flexibility. Appendix A to Part 100 was also removed.
- Changes to make 10 CFR 100.11 technology inclusive and more consistent with 10 CFR Part 100 Subpart B, Part 50, and Part 52.
- New definitions
  - *Tier 1 reactor* means a *power reactor* having an unmitigated consequence of less than 25 rem total effective dose equivalent at the site exclusion area boundary.
  - *Tier 2 reactor* means a *power reactor* having an unmitigated consequence of greater than 25 rem total effective dose equivalent at the site exclusion area boundary or where the unmitigated consequence is undetermined.

Related Guidance

- Revision 1 to RG 1.208 to address seismic siting for testing, microreactors, and power reactors by including graded approach
- Revision 2 to RG 1.23 to include a new appendix to provide details on how applicants can use offsite meteorological data sources for plant licensing
- DG-4036 on a graded approach to site characterization for all hazards and calculating nuclear siting category from unmitigated dose to determine Tier

DG-4036, Graded Approach to Site Characterization for New Reactor Applications

- Guidance on performing unmitigated consequence analysis to determine nuclear siting category
- Considerations for developing a site parameter envelope
- Graded Approach to Site Characterization
  - Use of existing site characterization information
  - Use of alternative site investigation methods
  - Appropriately scaled site investigations to support application development
- Appendix B of DG-4036 provides guidance on optimizing the seismic design basis

Determinate and Data-Backed Thresholds for Reactor Safety Assessments

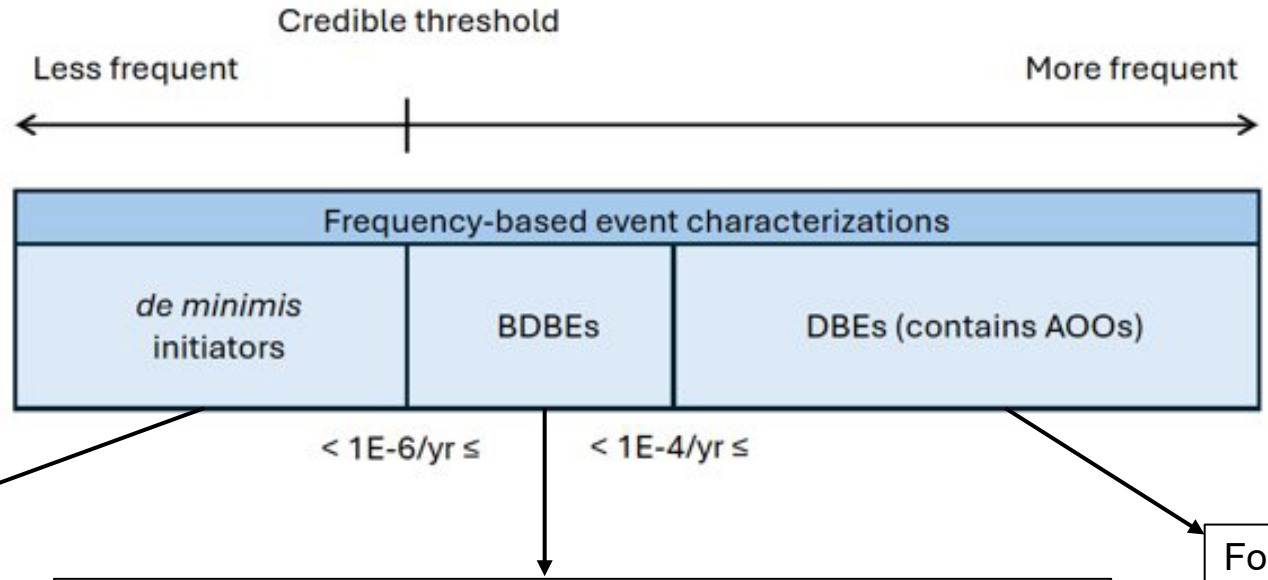
Background

- A significant portion of reactor safety assessments are done to assess the plant response to design basis events (DBEs) consistent with NUREG-0800 (Standard Review Plan (SRP)), Chapter 15, “Transient and Accident Analysis.”
  - The guidance in SRP Chapter 15 is qualitative and was developed from judgment and operating experience with large light-water reactors.
  - The current DBE spectrum does not reflect use of a determinate and data-backed threshold and does not fit well with evolutionary light-water reactor or advanced reactor designs.
- Introducing a definition of DBE into 10 CFR Part 50 that enables the use of thresholds would better support evolutionary light-water reactor and advanced reactor design and licensing processes by allowing removal of less frequent initiators from reactor safety assessments.

Proposed Changes

- Revise 10 CFR 50.2, “Definitions,” to include definitions of DBE and beyond design basis event (BDBE).
  - Definitions will establish that categorization as a DBE or BDBE is based on initiating event frequency thresholds.
  - The DBE category will capture credible events whose treatment will be consistent with current regulatory practice.
  - The BDBE category will capture credible events that fall below the DBE threshold using more realistic methods and will establish a frequency below which no regulatory treatment is necessary.
- Revise 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” to make a conforming change to the definition of DBE included in 10 CFR 50.49(b)(1)(ii).

Proposed Numerical Thresholds



No regulatory treatment (does not apply to events required to be evaluated by regulation).

Assume realistic conditions with consideration of uncertainty.
Assume all SSCs operational unless deemed inoperable due to the initiating event.
Calculated results may be based on best estimate methods, provided an evaluation of key uncertainties and cliff-edge effects is provided.

Follow traditional conservative methods (e.g., SRP Chapter 15; RG 1.203, “Transient and Accident Analysis Methods”).

Related Guidance

- DG-1454 (Draft RG 1.269), “Implementation of Determinate and Data-Backed Thresholds for Reactor Safety Assessments,” has been drafted to provide recommended numerical thresholds, graded methods (including qualitative methods) for determining event frequency, and information on design basis parameter selection.

Applicability and Expected Impact

- Applicability
  - Proposed definitions are intended to be a **voluntary** option for current commercial power reactor licensees and current applicants.
  - Proposed definitions are intended to be **mandatory** for future commercial power reactor applications submitted on or after 6 months after the effective rule date.
- Expected impact
  - Focus reactor safety assessments on credible, realistic risks within existing Part 50 and Part 52 frameworks by improving clarity and regulatory certainty for applicants and NRC staff.
  - Streamline licensing decisions and reduce the need to request exemptions.

Alternative Risk-Informed and Performance-Based Acceptance Criteria for All Part 50 and 52 Regulations

Alternative Risk-Informed and Performance-Based Acceptance Criteria for All Part 50 and 52 Regulations

- Affected regulations: 10 CFR 50.220 (new), 10 CFR 52.220 (new); 10 CFR Part 50 App A (GDC introduction & GDC 28)
- Creates a predictable process to pursue risk-informed and performance-based alternatives to acceptance criteria within 10 CFR Parts 50 and 52
- Lower entry threshold than an exemption – no special circumstances required
- Proposed guardrails compared to an exemption:
  - Higher safety threshold for approval (same level of safety) vs. exemption (no undue risk)
  - Applies only to acceptance criteria vs. any portion of the regulation
  - Leverages risk-informed criteria for justification
- Increases regulatory flexibility while maintaining appropriate NRC review and oversight
- Updates GDCs to support alternative analysis approaches

Risk-Informing 10 CFR 50.59 and Allowing Flexibility for Changes to Methods

10 CFR 50.59 – Risk-Informed Screening Criteria; Modernizing the Review of Modeling/Simulation

Background

- The intent of 10 CFR 50.59 is to allow licensees to make changes to the facility without a licensing action, provided acceptable levels of safety are maintained as documented in the final safety analysis report.
- The 1999 statements of consideration established that risk results would not be appropriate to evaluate a change to the facility under 10 CFR 50.59 without further rulemaking.
- *Departure from a method of evaluation* was narrowly defined to restrict the ability to make non-conservative changes to analytical margin without NRC review.

Proposed Changes

1. Allow use of quantitative results in risk-informed, probabilistic risk assessment (PRA)-based evaluations to show that an increase is not “more than minimal” under 10 CFR 50.59(c)(2)(i) and (ii).
2. Add an exception to 10 CFR 50.59(c)(2)(viii) permitting method-of-evaluation departures when supported by a documented verification, validation, and uncertainty quantification (VVUQ) process performed under a 10 CFR 50.221-compliant VVUQ program.

Risk-Informed Screening Criteria

Risk-Informed Evaluation Option

- Allows use of quantitative risk results to demonstrate “no more than a minimal increase” under § 50.59(c)(2)(i) and (ii).
- Use of quantitative results is permitted only if defense-in-depth and safety margins are maintained and documented.

PRA Scope and Quality Requirements

- PRA must cover all relevant internal and external hazards; exclusions must be justified.
- Licensee must identify NRC-approved applications addressing the adequacy of each relevant hazard model.

Conditions Where PRA Results Cannot Be Used

- PRA results cannot be used if a relevant hazard lacks an approved application that addressed the PRA model.
- PRA results cannot be used if there are any open facts and observations relevant to the change.

Documentation Expectations

- Must document screening basis, surrogates used, and justification for model screening decisions.
- Must document key uncertainties, assumptions, model updates, and peer reviews relevant to the change.

Risk Thresholds (Proposed in draft RG 1.187 revision)

- $\Delta CDF < \mathbf{5\times10^{-7}/yr}$ and $\Delta LERF < \mathbf{5\times10^{-8}/yr}$.
- Baseline CDF/LERF must remain **below RG 1.174 cumulative thresholds** [10^{-4} / 10^{-5}] after the change.

Modernizing the Review of Modeling/Simulation

What 10 CFR 50.221 Enables

- Provides a regulatory pathway for licensees to change methods of evaluation under 10 CFR 50.59 when the change is supported by an NRC-approved VVUQ program.
- Shifts licensing oversight from reviewing results to reviewing the credibility basis for the models that generate those results.

Core Concepts Defined in 10 CFR 50.221

- Credibility – whether a model can be trusted for a specific purpose.
- VVUQ – verification, validation, uncertainty quantification, and all other activities used to establish that credibility.
- Modeling and simulation (M&S) risk – the combination of the likelihoods that the M&S is incorrect or misleading and the consequences of relying on the incorrect or misleading result for some decision making purpose.

What a VVUQ Program Must Include

- VVUQ activities – What specific VVUQ tools you are using (e.g., code verification, solution verification, etc.).
- VVUQ assessments – The results of using those VVUQ tools.

This is not blazing a new trail, but more like turning a well-worn path into a road.

Next Steps

- Proposed rule published in the *Federal Register*
  - **Early July**, 45-day comment period
- NRC evaluates public comments and prepares final rule
  - Congressional Review Act determination and EO 12866 review
- Final rule published in the *Federal Register*
  - Estimated **12/29/26,** 30-day effective date

Contents of the Proposed Rule Package

- Primary Proposed Rule Documents
  - *Federal Register* Notice
  - Regulatory Analysis
  - Environmental Assessment

- Supporting Rule Documents
  - 22 Draft Regulatory Guides
  - 1 Interim Staff Guidance
  - 1 NUREG Update
  - Information Collections Supporting Statement and Burden Tables

Questions?

Abbreviations

AOO	anticipated operational occurrence
BDBE	beyond design basis event
CDF	core damage frequency
CFR	*Code of Federal Regulations*
DBE	design basis event
DG	draft regulatory guide
EO	Executive Order
GDC	General Design Criteria
LERF	large early release frequency
M&S	modeling and simulation
NRC	Nuclear Regulatory Commission
PRA	probabilistic risk assessment
QA	quality assurance
QMS	quality management system
RG	regulatory guide
SRP	Standard Review Plan
SSCs	structures, systems, and components
VVUQ	verification, validation, and uncertainty quantification