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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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REGULATORY RULEMAKING, POLICIES AND PRACTICES

SUBCOMMITTEE

+ + + + +

TUESDAY

JANUARY 14, 2025

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The Subcommittee met via Video/Teleconference, at 8:30 a.m. EST, David A. Petti, Chairman, presiding.

SUBCOMMITTEE MEMBERS:

DAVID A. PETTI, Chairman

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

VESNA B. DIMITRIJEVIC, Member

CRAIG D. HARRINGTON, Member

GREGORY H. HALNON, Member

WALTER L. KIRCHNER, Member

ROBERT P. MARTIN, Member

SCOTT P. PALMTAG, Member

THOMAS E. ROBERTS, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANTS:

DENNIS C. BLEY

STEPHEN P. SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

DEREK A. WIDMAYER

ALSO PRESENT:

LARRY J. BURKHART, ACRS/TSB

ANDERS F. GILBERTSON, NRR/DANU/UARP

WILLIAM D. RECKLEY, NRR/DANU/UARP

JESSE L. SEYMOUR, NRR/DRO/IOLB

NANETTE V. VALLIERE, NRR/DANU/UARP

MICHAEL J. WENTZ, NRR/DANU

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

8:30 a.m.

MEMBER HALNON: Okay, it's 8:30. The chairman for the subcommittee is online, Dave Petti.

CHAIRMAN PETTI: Thank you, Greg. The meeting will now come to order. This is a meeting of the Regulatory Rulemaking, Policies and Practices Subcommittee of the Advisory Committee on Reactor Safeguards.

I'm David Petti, chair of today's subcommittee meeting. ACRS members in attendance in person are Ron Ballinger, Greg Halnon, Craig Harrington, Robert Martin, Scott Palmtag, and Tom Roberts. The ACRS members in attendance virtually via Teams are Vesna Dimitrijevic, Matt Sunseri, Vicki Bier, Walt Kirchner, and myself.

We have two of our consultants participating, Dennis Bley remotely, and Steve Schultz in person.

If I've missed anyone, either ACRS members or consultants, please speak up now.

MEMBER HALNON: Hey, Dave, at this point, sorry, we don't have Dennis on just yet.

CHAIRMAN PETTI: Oh, okay.

MEMBER HALNON: That's the only

correction.

CHAIRMAN PETTI: Thanks. Derek Widmayer of the ACRS staff is the Designated Federal Officer for this meeting. No member conflicts of interest were identified for today's meeting. We have a quorum for today's meeting.

During the meeting today, the subcommittee will receive a briefing on updates made to 10 CFR Part 53 entitled Risk Informed Technology Inclusive Regulatory Framework for Commercial Nuclear Plants. The updates were mandated by the Commission following its review of the proposed rule. The ACRS is statutorily required to review all safety-related regulations developed by the staff and this is the most significant addition to nuclear safety rules done by the staff in many years. The ACRS has held 17 subcommittee and full committee meetings with the staff as they developed the rule language that went to the Commission for its review. And this meeting is intended to familiarize the committee on the revisions made to the rule language as a result of the Commission's review. The committee does not plan on writing a letter report after this update.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, or

FACA. The NRC implements FACA in accordance with its regulations. Per these regulations and the committee's bylaws, the ACRS speaks only through its published letter reports. All member comments should be regarded as only the individual opinion of that member and not a committee position.

All relevant information related to ACRS activities such as letters, rules for meeting participation, and transcripts are located on the NRC public website and can easily be found by typing About Us ACRS in the search field on NRC's home page.

The ACRS, consistent with the agency's value of public transparency and regulation of nuclear facilities provides opportunity for public input and comment during our proceedings. We have received no written statements or requests to make an oral statement from the public. And we have also set aside time at the end of the meeting for public comments.

The ACRS will gather information, analyze relevant issues and facts, and formulate proposed conclusions and recommendations, as appropriate, for deliberation by the full committee.

A transcript of the meeting is being kept and will be posted on our website.

When addressing the subcommittee, the

participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. When you are not speaking, please mute your computer on Teams or by pressing *6 if you are on the phone. Please do not use the Teams chat feature to conduct sidebar discussions related to the presentations. Rather, limit use of the meeting chat function to report IT problems.

For everyone in the room, please put all your electronic devices on the 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, the table microphones are unidirectional and you will need to speak into the front of the microphone to be heard.

Finally, if you have any feedback for the ACRS about today's meeting, we encourage you to fill out the public meeting feedback form on the NRC's website.

We'll now proceed with the meeting and I'll call on Mike Wentzel from the Office of NRR for Opening remarks.

MEMBER HALNON: Hey, Dave, just real quick. This is Greg. Dennis has joined us, we're

still waiting on Matt Sunseri so those are the two corrections to the attendance. And just remind everybody, the court reporter needs to know your name so if you're talking and want to make a comment, please start with your name. Good ahead.

MR. WENTZEL: Good morning. My name is Mike Wentzel. I'm Chief of the Advanced Reactor Policy Branch in the Division of Advanced Reactors and Nonpower Production and Utilization Facilities in the Office of Nuclear Reactor Regulation.

MEMBER HALNON: Mike, you need to speak up. Get a little bit closer. There you go. That's good. Thank you.

MR. WENTZEL: So as was mentioned, we're here today to continue the staff's discussion with the committee on the rulemaking for a Risk Informed, Technology Inclusive Regulatory Framework for Advanced Reactors, better known as Part 53. As a reminder, this rule was developed in response to the Nuclear Energy Innovation and Modernization Act and provides an alternative framework for licensing inter-commercial nuclear plants that is technology inclusive, uses risk-informed and performance-based techniques, provides an equivalent level of safety to that of operating commercial and nuclear plants and

provides flexibility for licensing and regulating (audio interference) of technologies and design.

Our last meeting with the committee was in the fall of 2022 and significant milestones have been achieved since then. This includes the Commission's approval of the proposed rule with revisions for publication last March and publication of the proposed rule for public comment last October.

The initial 60-day public comment period was extended to February 28th based on several requests for stakeholders. The staff has held two public meetings since publication of the proposed rule. The first was in November in last year and the second one was just last week.

The purpose of today's informational briefing is for the staff to provide an overview of the changes that were made to the draft proposed rule as a result of the direction from the Commission. We will also provide some background material and discuss key provisions of the proposed rule as a refresher and potential aid to newer members of the committee.

And finally, we intend to discuss a draft white paper on potential revisions for testing of fuel manufactured reactors in the manufacturing facility. This draft white paper was publicly released last

month and discussed with stakeholders during their public meeting last week in response to direction from the Commission.

Because the public comment period does not close for another six weeks, the staff will not be discussing any public comments received to date, nor can we discuss the staff's thoughts about potential changes to the final rule at this time. We will work with the ACRS staff to schedule future interactions with the committee between the close of the comment period and the delivery of the draft final rule to the Commission currently scheduled for no later than May 1st, 2026. And as a continued reminder, we do appreciate the past interactions with the committee on this important rule making and we're really looking forward to the discussions today and your reactions to the report. Thank you very much.

MEMBER HALNON: Okay, so go ahead, Anders.

MR. GILBERTSON: Thank you, good morning, everyone. My name is Anders Gilbertson. I'm the senior project manager in the Office of Nuclear Reactor Regulation. I'm one of the technical leads for the key NRC staff.

Today, a lot of the opening remarks here have already addressed some of the points on the first

few slides, so you can go to the next slide, please, Slide 2.

Today, I'll be presenting with Bill Reckley and Jesse Seymour will be touching on different parts of the rules here (audio interference). I will be going through the majority of the first portion of the presentation and then I'll call on Jesse and then I'll close things out.

Next slide, please.

Okay, and so this is just a reference slide to provide some hyperlinks to the proposed Part 53 rule published on October 31st of 2024 and the various places it can be accessed, as well as associated documents, guidance documents and such. The explanation of what those documents are can be found in Section 19 of the FRN under the Availability of Documents. And then the white paper that Bill will be touching on, Bill Reckley will be touching on later, that can be sourced through the ADAMS accession number.

And of course, just to reiterate that the past ACRS interactions have been very informative and instrumental to the formation and development of the proposed Part 53 rule. So we greatly appreciate those interactions.

Next slide, please.

And again, another sort of set of references, just the SECY paper that was delivered to the Commission March 1st that provided the draft proposed rule and then the SRM that was approved and the proposed rule with exceptions and clarifications. And of course, we'll be talking about that relative to direction that was given to the staff and the SRM and the differences between the draft proposed rule and the published proposed rule.

Next slide, please. Slide 5.

Okay, and so again, this is just to reiterate the sort of timing that we're talking about here, so the public comment period closes at the end of February. We'll look forward to our subsequent ACRS interactions between the end of that period and when we start our internal concurrence May 2026. May 1st is when we'll be sending the schedule to send the rule, the draft final rule to the Commission, and then of course, the NEIMA deadline is the end of December 2027.

Next slide please. Slide 6.

Okay, so I'll get into a little bit of background here just to set the stage where we are. So this diagram is just provided to help illustrate

that there have been numerous activities that the NRC has undertaken to address involving stakeholder needs and prepare for licensing and oversight of advanced reactors.

As was previously mentioned, the Part 53 rulemaking was directed by Congress through the Nuclear Energy Innovation and Modernization Act and the foundations of the proposed rule are built on Commission rules, policies, and decisions on risk-informed and performance-based regulations, as well as lessons learned from experience with the Part 50 and Part 52 regulatory frameworks. And the Part 53 proposed rule regulatory framework directly incorporates the use of risk analyses to inform identification of licensing-basis events and safety classification of plant equipment and it leverages performed-based approaches to enhance aspects such as staffing flexibility and achieving adequate defense in depth. So in that regard, the Part 53 proposed regulatory framework would call for designers and operators to enhance flexibility while allowing different proposed approaches to satisfying high level safety criteria versus meeting more prescriptive requirements under the Parts 50 and 52 regulatory frameworks.

Slide 7, please.

To give a relatively brief, high-level overview of the Part 53 draft proposed rule, it consisted of two independent frameworks that were referred to as Framework A and Framework B. So broadly speaking, Framework A was developed to support a top-down approach for developing a safety case where an applicant would meet high level safety criteria by performing analyses to define necessary safety functions which, in turn, are fulfilled by design features that lead to specified functional design criteria identified in the analysis. So this framework would feature probabilistic risk assessment, or PRA, a key analysis tool for systematically analyzing the risk of design for normal operations.

Framework B was developed to support the bottom-up approach for developing the safety case. It was based on adapting established design criteria, design rules, and deterministic analyses approaches to accommodate a wider range of reactor technologies. And Framework B would have provided for more traditional confirmatory uses of risk insights from PRA or an alternative evaluation of risk insights approach which has commonly been referred to as AERI.

MEMBER HALNON: Dave just raised his hand.

MR. GILBERTSON: Yes.

CHAIRMAN PETTI: Anders, just a question of AERI. You know in our letters we're big fans of that approach, particularly for potentially for microreactors. Given that it still exists as a draft Reg. Guide, but it doesn't sound like there's plans to turn it into a Reg. Guide, what status does that mean for a licensee that might want to use it? Does it always stay draft forever if it never gets evolved? Is there a time line that it just evaporates or something? How does that work?

MR. GILBERTSON: So as I understand it, when the staff published draft guidance, my understanding that doesn't get withdrawn from the public domain. It exists even if it's not formally converted to a final regulatory guide. So I suppose, in principle, that could be taken, it could be adapted perhaps, understanding that, of course, it's very specific to what was formally referred to as Framework B and the requirements that had been proposed therein.

CHAIRMAN PETTI: Okay. Thanks.

MS. VALLIERE: Excuse me. This is Nan Valliere on the Advanced Reactor Policy Branch. And I just wanted to add a couple thoughts on this response to the next question. So the first is that

recall, all of you were here, recall the Commission directed us to remove Framework B from the rule which is where AERI was cited in the draft final rule. However, they also directed the staff to go forward with an options paper on how to implement a Framework B approach going forward and they specifically directed the staff to include AERI in all of the options. So it is not -- not quite that yet I guess I would say.

(Laughter.)

The second point I wanted to make is that it is the AERI guidance document, because it was removed from the proposed rule, that draft guide was not published for public comment with this rule. So it has never even gone out as a draft guide. I just wanted to provide that for everyone.

MEMBER HALNON: This is Greg. We did see it as a draft guide though from an ACRS perspective, so it's in the public domain.

Dennis?

DR. BLEY: Yes. Nan, what's the status of that paper you said you were doing -- to put together?

MS. VALLIERE: Yes, it is due to the Commission in March and we are actually having a public meeting on it later this week. So it's due

shortly.

DR. BLEY: Okay. Will you be bringing that to the committee?

MS. VALLIERE: Not at the options stage, but I think once the Commission directs us which option to go forward with, I think that would probably be more the likely time that we would engage with the committee.

DR. BLEY: Okay, thanks.

MEMBER MARTIN: This is Bob Martin. Continuing on this theme option, is that potential evolution of Part 53 or maybe in the 52-50 realm?

MS. VALLIERE: Yes, so the options that the Commission discussed basically were added to Part 50, create a new part, or create a very high-level rule, but most of what had been in the original framework into guidance.

MEMBER HALNON: Go ahead, Anders.

MR. GILBERTSON: All right, thanks. Okay, so -- again, this Anders Gilbertson. So Framework B, just to finish off this slide, it also included requirements to develop and use principal design criteria similar to the general design criteria and Appendix A to Part 50 and would have provided for technology-inclusive approaches to meeting existing

requirements developed for all 50 parts. So just to cap that off and of course, we have emphasized that there will be a subsequent meeting later this week, public meeting on that options paper.

Slide 8, please.

Okay, so we want to start to get into the SRM direction. The Commission included a total of 15 items shown here from the staff to address in the Part 53 proposed rule before its publication comment. In the interest of time, what we are planning to do is really just focus on the key items from the SRM that were more complex to address, required a little more depth of thought and are highlighted here on the list, items 1 through 4 and 8.

And like Mike had mentioned earlier, this will be set in the context of a broader presentation of all of the subparts of the rule at a very high level. Again, this is mostly just to express and communicate what the updates are to the proposed rule. So we'll touch on a few of these other SRM items, but many of them were very straight forward to simply just making the more clerical changes involved.

On subsequent slides, we have some purple-colored boxes at the bottom just to help orient you to what some of those key changes are based on the

implementation of the SRM and to help provide a general roadmap of those key changes.

Slide 9, please.

Okay, so this table just shows the general organization of the Part 53 proposed rule relative to the subparts of the rule text, going into a little more detail on the next slide. A couple of the key SRM items to note on this slide straight away are that consistent with the first SRM item, the collection of requirements referred to as Framework B were removed entirely from the proposed rule and other rulemaking documents as we just touched on. And then consistent with the SRM item number ten, the draft proposed requirements in subpart K related to quality control were also removed and replaced with references to the existing quality assurance requirements under Appendix B to 10 CFR Part 50. So that's why subpart K is not shown on this list. And again, both of these were -- they're fairly impactful to the rule. They will go straight forward to implement.

Slide 10, please.

Okay, so this slide goes into a little more detail on the structure of the rule and this is set in the context of the overall project life cycle as it would generally be sequenced through the process

of licensing at a facility. This writing on the upper left, that's subpart B would provide the high-level technology, includes the safety criteria. Those would serve as foundational performance standards for the subsequent performance-based requirements in the other subparts. And those other subparts would address how specific activities during various stages of the facility life cycle would contribute to satisfying the high level performance standards.

Now the performance standards in subpart B would also establish a means to determine appropriate regulatory controls for SSCs, human actions, and programs. And likewise, moving on to the other portions of the diagram, subparts D, E, and F would provide requirements related to consideration of siting issues to construction and manufacturing and operations, respectively. And of course, all as it relates to meeting safety criteria that are defined in subpart B.

Subpart G would provide requirements for plant decommissioning activities and license termination and subparts H and I would provide requirements for information related to license certification approvals and the maintenance of licensing basis information.

And then toward the bottom here, subpart A would provide the general provisions of the proposed rule, in particular, includes terms and definitions as they would apply for the proposed Part 53.

Subpart J would provide administrative and reporting requirements for the entire life cycle and subpart M would provide enforcement requirements.

So it's important to keep in mind that the Part 53 proposed rule incorporates various concepts from the current regulatory frameworks and the licensing modernization project methodology in a technology-inclusive, cohesive, and efficient manner. As such, the concepts that would be incorporated into the Part 53 proposed rule were integrated across and serve as foundations for why different aspects of the regulatory framework develops the way they were and we will point out many of these areas as we go through this presentation.

All right, moving on to Slide 11, please.

Okay, so getting into -- from this point forward, we'll be getting into the substance of the rule and touching on some of the SRM items, some of the more detailed ones.

So subpart A is general provisions for Part 53. It would be applicable to all applicants and

licensees and these requirements would be largely equivalent to the related general provisions under Part 50. More specifically, for example, the proposed Sections 53.40 through 53.120 are equivalent to their related requirements under Part 50. And general differences between the proposed Part 53 and Part 53 frameworks include framework-specific references to other portions of the Part 53 regulations versus references to Part 50 regulations, of course, as well as the definitions that are specific to Part 53.

Slide 12.

DR. BLEY: Anders?

MR. GILBERTSON: Yes.

DR. BLEY: Dennis Bley. You said I think 40 through 120 were equivalent to those under 50. Are they identical or just supposed to be equivalent?

MR. GILBERTSON: They are -- I believe they are nearly identical. I know Bill Reckley could certainly publicly say more specifically off the top of his head, but I believe they're almost identical.

DR. BLEY: Okay, because if they're identical that raises a question to the possibility of confusion occurring if they're meant to be the same. It's just a concern.

MR. GILBERTSON: Okay. Understood. Thank

you.

Slide 12, please.

Okay, so the matter of fostering clarity and consistency regarding the use of defined terms. Most of the definitions under the proposed 50.20 -- sorry, 53.20 would be equivalent to definitions of corresponding terms defined under 10 CFR 50.2, 10 CFR 52.1, and other existing NRC regulations. Also, NEI 18-04, Provision 1, as that's endorsed by Reg. Guide 1.233, Revision 0. Of course, both of those are the documents that relate to the licensing modernizing project methodology and definitions corresponding to those in the ASME/ANS consensus PRA standard for non-light water reactors. That's designated as ASME/ANS RA-S-1.4-2021, and of course, as that's endorsed in Reg. Guide 1.247.

Okay, so next slide, Slide 13, please.

So that's the general summary of subpart A. So getting into the technical requirements here, subpart B would provide the technology inclusive safety criteria that again would serve as the performance standards for the subsequent performance-based requirements used throughout Part 53.

As they talked about these proposed requirements under subpart B, I'll also touch on some

of the foundational concepts that form the basis for the proposed requirements on the next slide.

Slide 14, please.

MEMBER MARTIN: Wait.

MR. GILBERTSON: Yes.

MEMBER MARTIN: This is Bob Martin. In the earlier drafts some critique public comment related to expansion, potential expansion of ALARA. Are you going to address the evolution of ALARA as safety criteria in your presentation?

MR. GILBERTSON: Yes. Yes. That is one of the items. Those are items that the Commission directed the Staff, so I'll, yes, I'll be talking about that in a few slides here.

Okay, Slide 14. Here we go. Okay, so this figure, as you might be familiar, the central portion of the figure comes directly from Regulatory Guide 1.174. And this is to help provide a visual representation of how the different portions of the Part 53 proposed rule would relate to the principles of integrated risk-informed decision making described in Reg Guide 1.174. And that would be foundational to NRC determinations of reasonable assurance of adequate protection under the Part 53 proposed rule.

So although the principles in Reg 1.174

are framed in terms of risk-informed licensing basis changes for an operating reactor, specifically a light water reactor, those principles can readily be generalized to any risk-informed decision making. And they have in many different contents. So this is natural that it was brought into Part 53.

And of course, the use of plant risk under the Part 53 proposed rule would be one of several performance measures used in the subpart B. The proposed rule would also use multiple performance standards related to deterministic criteria and defense-in-depth measures.

So in that way NRC's approval of using a comprehensive risk metrics, metric, or sets of metrics with associated risk performance objectives would not be, by itself, an indicator of adequate protection. They would be one piece of a suite of regulatory requirements that when consider holistically would inform the basis for NRC decision.

And this approach is analogous to that used for plants licensed under Part 50 and Part 52 in that no single regulatory requirement governs whether a facility is determined to be safe enough.

DR. SCHULTZ: Anders, this is Dave Schultz.

MR. GILBERTSON: Yes.

DR. SCHULTZ: Is this the way in which the Staff has responded to the Commission's concern about allowing flexibility in PRA acceptability determinations or is that, you're going to talk about that later?

It seems like it might be associated with this description?

MR. GILBERTSON: Yes, I will talk about that later. I have a slide on that. We don't, let's see. It's certainly related to these principles, but of course because it is one of the principles, a risk-informed, you know, consideration of risk insights and such. So yes. I mean, I it did inform that in part I would say.

DR. SCHULTZ: Okay. I guess what I'm looking for too, you deleted Framework B, and then I looked at Framework A and looked at the various descriptions associated with PRA. And I was looking for something that would in fact address the Commissions concern here. And I didn't see changes that reflected a movement toward addressing the concern.

And perhaps there is other documentation that's going to be utilized in terms of reg guides and

so forth. But within the rule itself I didn't see changes that address the Commissions concern.

MR. GILBERTSON: So specifically are you, you are referring to the general concern that risk, consideration of risk insights is just one piece of the overall decision making process?

DR. SCHULTZ: That's right.

MR. GILBERTSON: Okay. Okay. So --

DR. SCHULTZ: I really didn't see, I looked at everything associated with PRA in Framework A. I didn't see changes that affected the Commission's concern. And perhaps -- that's why I pointed it out here. You got a lot of different features associated with guidance that might address the concern but it's not within, directly stated within the Framework A, as far as I could see.

MR. GILBERTSON: Right. So I guess maybe what I would offer, and I'll also talk about it a little bit more when I get to that point is, I guess the more general characterization of the rule that it's risk-informed versus risk-based, that's something that we've tried to focus on and make sure that we're staying true to that notion.

Risk-based would be very different. I think a different framework. We would have written

requirements differently to address something where you are making decisions purely based on these insights. Purely based on what the probabilistic risk assessment is telling you.

My personal view is that requires a much higher degree of pedigree of the PRA and such to be able to even entertain that kind of decision making. So I think it's inherent in the rule in the way you've written it.

That point, you know, it is more embodied in the guidance, which I think is why, for example, for SRM Item 3 we opted to adjust the preamble and not the rule language. We thought it was more appropriate to talk about what those existed processes were and explaining more about what pure acceptability is as it currently stands in the practice.

DR. SCHULTZ: Thank you.

MR. GILBERTSON: Okay. Okay, we can move on to Slide 15. Okay, so going now to the safety criteria themselves we'll start with the proposed 53.210. We don't have any highlights on here, any purple boxes, so again, this is going to be one of those areas where I'm going to talk about this at a higher level just to help set the context and orient us to what this framework is doing.

So the Part 53 proposed rule would maintain an important role for deterministic analyses of design basis accidents in the performance of criteria for the proposed 53.210 and the related analytical requirements provided in proposed 53.450(f).

So starting with the proposed 53.210. This would provide a DBA safety criteria analogous to the DBA, requirements for DBAs under 10 CFR 50.34(a)(1)(ii)(d) as it relates to the 25 rem reference value for a potential radiological consequences and other similar requirements under Part 52 and Part 100.

And SSCs relied upon to demonstrate compliance with the criteria in 53.210 would be classified as safety related. And the use of safety related SSCs, and the 25 rem reference values for potential radiological consequences, would align with traditional deterministic approaches for LWRs from Parts 50.34, 52.79 and 100.11 for evaluating the effectiveness of plant design features relative to possibly the reactor absence.

MEMBER HALNON: Vicki Bier has a question.

MR. GILBERTSON: Go ahead.

MEMBER BIER: Thanks. In the traditional

regulatory approaches there was always this contradiction, not contradiction but paradox maybe, of things that were safety significant but not safety related if they were in the secondary systems. And would that not be the case with this new definition of what's safety related?

MR. GILBERTSON: I guess I would say not necessarily because of, yes, I'll get into this on the next slide as well when we talk about the safety criteria for licensing basis events other than the DBAs. Because the analyses that supports meeting those safety criteria includes a broad survey of the spectrum of risks that a facility might be exposed to, that provides, would provide much more information than I think some of the more prescriptive sets of assumptions under Parts 50 and 52.

So it's -- the definition of safety related under Part 53, it stems largely from what your -- what the design basis accidents are showing and demonstrating that you can meet the safety criteria, but it's also underpinned by those other analyses for the licensing basis events other than design basis accidents. So I'm not sure if I answered your question, but.

MEMBER BIER: Well, I think it was

actually a pretty clear explanation of why something might be safety related or not. You know, it sounds like it won't eliminate the possibility of that situation where something is safety significant for non-design basis, beyond design basis events, but not safety related.

So in a way it's kind of too bad that it doesn't resolve that, but I think the explanation of what it's doing is pretty clear. Thanks.

MR. GILBERTSON: Okay. Yes. And maybe I would just offer to finish that thought. That the underlying analysis using the PRA perhaps would offer more confidence that for those items that are designated as non-safety related but safety significance, we would have a better understanding of that, and confidence in those designations because they're underpinned by, again, a systematic analysis provided by the PRA.

MEMBER ROBERTS: Okay. Anders, this is Tom Roberts. There were no AOOs in this scheme? Are AOOs a subset of DBAs in this definition?

MR. GILBERTSON: AOOs would be a category of licensing basis event. I'll actually get into that, I think maybe in a couple of slides. I provide the definition of what licensing basis events are.

And yes, that's one of the categories. Along with DBAs -- and I'm forgetting off the top of my head the other, but there are four, four categories. And we'll touch on that on a later slide.

MEMBER ROBERTS: Okay. Maybe it's the next slide it will come up. But other than DBAs, I've confused maybe, because that would seem to include both AOOs and beyond design basis events. It seems odd to have a category that includes both of those.

MR. GILBERTSON: Yes. So, and we can talk about that. You know, the designation of essentially, you know, LBE being sort of a design basis accident, or not a design basis accident, it really relates to this dichotomy of the safety criteria, how they're defined for DBAs and for, I'll say, non-DBAs just to latch that down a little bit more.

But yes. So the AOOs wouldn't be a class of DBAs. The AOOs, again, I'm forgetting the next --

PARTICIPANT: Unlikely event sequence.

MR. GILBERTSON: -- unlikely, yes, unlikely event sequences and the very unlikely event sequences. I'm sorry, we use different terms in Part 53 than we do for LMPs so I trip over that sometimes.

But those three categories are their own categories. Those are informed by the PRA analysis.

The DBA is a separate category that is informed by the PRA analysis but it serves the deterministic piece of the Part 53.

MEMBER ROBERTS: Yes, maybe it will become clear on the next couple of slides. It seems odd that an AOO, if you had SSCs that required to mitigate AOOs those would not be safety related but they would be if it was to mitigate a DBA. So an SCC that mitigates a more than likely scenario, being, you know, subject to less quality standards it seems odd to me. Maybe I don't understand how the DBA and other DBAs would apply.

MR. GILBERTSON: Okay. Yes. Maybe we will, we can, we'll probably get to that on a later slide. But yes, certainly if it's not coming back up I expect you'll ask again, so.

Okay. So going back to 53.210. So like I mentioned, the requirements there for the 25 rem reference values are similar to what you would see in Parts 50, 52 and 100. And as such we included a footnote in the proposed 53.210 to similarly explain that the use of the 25 rem reference value would not be intended to imply that it constitutes an acceptable limit for an emergency dose to the public under accident conditions. But it is in fact just a

reference number that is used in evaluating plant design features with respect to the DBA case.

And the inclusion of the safety criteria for DBAs in subpart B would provide a lot of structure supporting the identification and treatment of safety related SSCs and establishing the corresponding functional design criteria for those SSCs.

MEMBER ROBERTS: And, Anders, you're saying that that criteria of 25 rem associated with design basis accidents would be across the board for all design basis accidents?

MR. GILBERTSON: That's correct. Yes.

MEMBER ROBERTS: Thank you.

MR. GILBERTSON: Okay. So, going to the analysis. The DBAs analyzed under Part 53 would be similar to the traditional DBAs analyzed under Parts 50 and 52. However, Part 53, the DBA analysis would be more narrowly focused on selecting safety related SSCs and determining functional design criteria for those SSCs to ensure that a facility conservatively meets the safety criteria in 53.210.

So the overall control of risks posed by a commercial nuclear plant under the Part 53 proposed rule would be provided by the analysis of and measures taken for both DBAs and other LBEs, including the very

unlikely event sequences. And so the analysis of DBAs under the proposed 53.450(f) would be required to address event sequences derived from those with estimated frequencies below the expected lifetime of a generation of reactors. So for example, event sequences with frequencies as low as one in 10,000 years.

And as it's proposed in 53.450(f), DBAs would need to be analyzed using deterministic methods and ensure a safe, stable end state and only rely on safety related SSCs. And if needed, human actions performed by operators that would be licensed under the provisions of Sections 53.760 through 53.795. Jesse Seymour will be speaking about this later this morning.

Now to draw a contrast to how DBAs are analyzed under Part 50, the analysis of DBAs under Part 50 are used to provide bounding assessments, incorporate standard design rules, such as assumptions related to single failures, and to define conservative performance requirements for safety related SSCs. So limitations related to the traditional deterministic approach were addressed in Part 50 through case-by-case assessments. And specific actions for beyond design basis events such as anticipated transients

without SCRAM and station blackout. Which the proposed Part 53 rule has been designed to avoid. The limitations that is.

Slide 16 please.

MEMBER KIRCHNER: Anders, this is Walt Kirchner. Before you go on --

MR. GILBERTSON: Yes.

MEMBER KIRCHNER: Have you kind of done a mental or a tabletop equivalent exercise that using 53 would demonstrate a equivalent level of safety to the assumptions used in 50 or 52 with regard to DBA analyses?

And I'm thinking specifically of assumptions like single failure, control room, maximum worth control rod out or in, et cetera. Have you gone through that for a tabletop exercise to actually demonstrate that you could say you're providing an equivalent level of safety?

MR. GILBERTSON: So I will take a shot at answering that but I'll probably have to defer to either Bill or Nan because they have a lot more of the history.

Don't know that there were any specific tabletop exercises done. The thinking that went behind that was, I believe at more sort of conceptual

level, and part of the integration of the PRA into the Part 53 requirements, together with the analysis of DBAs, those taken as a whole are sort, you know, provide a foundation for the staff being able to make those kind of conclusions that there is an equivalence of level of safety.

As I discussed before because, you know, in part the PRA provides such a systematic understanding and interrogation of how a facility would perform under those conditions. Bill or Nan, would you offer any other thoughts to that point?

MR. RECKLEY: Yes. Anders this is Bill Reckley. Walt, I would say we're confident that in total we get there. But as we talked about in the preamble the role of the design basis accident is a little different in Part 53. As Anders mentioned, more narrowly focused to define the design requirements, performance requirements of safety related SSCs whereas in Part 50 there is kind of a bounding event kind of role for the DBA.

So in total, and Anders is going to get to the licensing basis events other than the DBA in the next few slides, in total we're confident that there is an equivalent level of safety. But I would just caution and refer you to the preamble discussion of

the DBA because it, itself, serves a slightly different role.

MEMBER KIRCHNER: Okay, thank you.

MEMBER MARTIN: Okay, Bob Martin. Kind of in the spirit of Walt's question, you know, I see, we have this nice track changes version and I can see what has been and has since been removed. I see beyond design basis accidents and a lot of editing associated with that and elevating the emphasis on DBA's course, traditionally.

We've had Chapter 19 where deterministic, beyond design basis accidents have been formed under the best estimate sense. They do contribute specifically in a defense-in-depth context.

I'm trying to wrap my head around whether we have equivalence with Part 53 in that sense. I mean, do deterministic beyond design basis accident analysis go away or what does it look like in this world?

MR. GILBERTSON: No, they wouldn't go away they would be captured by the class of LBEs that are the very unlikely events. So those would be the event sequences that come out of the risk, the PRA and other approaches and risk analyses to show what the more extreme conditions that might be, a facility might be

exposed to.

And like you said, those are informing decisions from the start of the design to inform defense-in-depth consideration, whether it's adequate defense-in-depth. Which can allow flexibility to make decisions about adding additional capability to a facility to address particular conditions which may or may not be performed or satisfied by SSCs that aren't necessarily safety related, but you have extra layers of defense. Whether it's physical equipment, programmatic controls or operator actions.

MEMBER MARTIN: And now in the vision for, you know, a Part 50, well, really not just Part 53 SARs we have like a 7 Chapter, or whatever, as opposed to the traditional 19 Chapter SAR. And now, all of these analyses, ones that land in three or, anyway. One of those early chapters.

And I'm trying to, you know, think, well now, we have an emphasis on design basis accidents primarily for the purpose of demonstrating the performance of safety related SSCs, and then we have, as you just mentioned, the LBEs which might, which of course include the DBAs, but get into the story of defense-in-depth. I mean, does this kind of appear in segregated sections there?

I mean, just like Chapter 19 is separate from Chapter 15. I mean, do we, is there intention to still have a holistic integration of all these analyses or do we still kind of separate that out to tell two different stories. One to support the safety related aspects of the design and another to tell the defense-in-depth story?

I mean, I'm not going to judge it one way or the other. I'm leaning towards, you know, kind of the old way of segregating because it's much more transparent. Or is the idea to integrate the stuff and then somewhere in there you can pick up both stories.

MR. GILBERTSON: Right. Right. Yes, and so, I think that it's -- the explanation will be more of a holistic, integrated explanation. We're, you know, part of how we're seeing that is as it relates to the guidance that we provided on content of applications for non-LWR applications using LMP under Parts 50 and 52. Which we refer to as the ARCAP/TICAP guidance. Which I think is a structure that you're referring to. Which relates to NEI 21-07 and how those chapters are laid out.

So I think it's kind of maybe a combination of both of those things. There are, you

know, there needs to be an understanding of how defense adequate defense-in-depth is achieved. Whether that's through capabilities, programs, et cetera.

But there is a sort of narrative I think that needs to be put together to explain how that relates to the other pieces of the process. Talking about ARCAP/TICAP, the process of LMP as you're talking about how you perform your licensing basis even identification. How you did your structures system component safety categorization.

MEMBER MARTIN: Right.

MR. GILBERTSON: Because all of that has an interplay with defense-in-depth. They all relate to each other. And of course, that's, it's a very iterative process. So.

MEMBER MARTIN: Sure. I worry that maybe the edits that made, you certainly have elevated the intention on DBAs but the defense-in-depth story almost takes a backseat to that where they really have to be, you know, again, in a holistic view of these things equivalent. At least in the articulation. And it's all very hard too.

But I just was worried a little bit, but de-emphasizing of the importance of the defense-in-

depth story and the beyond design basis, accidents, you know, that maybe in some way there was less burden, not to say there should be more, but to Walt's comment about equivalence. You know, making it, you know, easier because it remains, I would say, to more important in the kind of technology inclusive approach to have this holistic view of these things.

And here we are kind of emphasizing traditional terms where I, you know, I think we had a lot of support for NEI 18-06 and 21-07. And this seems to be a little bit of a pull back to the way things have been done. I mean, from what you had in the original draft.

MR. GILBERTSON: So I guess --

CHAIRMAN PETTI: This is Dave. Let me just, let me try something here. My view is that the defense-in-depth under 53 and LMP takes a more balanced view across the spectrum, the frequency spectrum, where you don't necessarily know a priori where your two risks lie in a deterministic world. And it's a method to make sure that it's applied in a balance way so that you get the proper safety, quote-unquote, whatever that means.

And so that, to me, that's kind of like the building block of the whole, of Part 53 and LMP.

And then you breakout this DBA role where you only assume, you know, the safety systems work. And that's just to demonstrate that they're doing their job.

And so the defense-in-depth is built in from the beginning. Now it actually gets articulated you won't know until we see the first application. But that's always how I can remember it being discussed in the earliest days when we had, for instance, some of the guys who actually put LMP together come in and talked with the committee.

MEMBER MARTIN: Well I absolutely agree, Dave. I'm just kind of reacting to the changes in the, from proposed to what we're seeing here today and seeing a lot of removal of the content that, you know, emphasized a beyond a design basis kind of on par with everything else. Or really in the context of licensing basis events.

Now, I just kind of worry this may be an overreaction to comments that maybe, you know, beyond design basis events is bleeding too much into design basis. Where we are, you know, going back to Walt's comment, it's about equivalence. We're always looked at beyond design basis events but, you know, how this gets interpreted by an applicant remains to be seen.

And I think the, if I think about an NE

18-06 model, and I have some experience doing that, I can, you know, integrate a holistic type approach with it. And then in the synthesis of the analyses answer two questions. One is, what is the performance of the safety related equipment that you care most about, and then what is the defense-in-depth story.

But I would hate to think that someone, okay, well let's just focus on design basis events and then we'll throw in a few extra events here just because it's mentioned. Again, we won't know until at least the filing.

MEMBER HALNON: You're talking about 18-04 right?

MEMBER MARTIN: Yes. Yes, yes.

MEMBER HALNON: I have experience with 18-06 translates into --

MEMBER MARTIN: Oh, 18-04, I am so sorry.

MEMBER HALNON: All right --

MEMBER MARTIN: That's a different, yes, 18-04.

MEMBER HALNON: So this is Greg. I just wanted to, I've been thinking about this equivalent stuff and you mentioned SBO or station blackout. Is that, I'm thinking you can make a qualitative argument that it's even more likely you could have a station

blackout with a new reactors because of less emphasis on, less emphasis on switch yards and redundancies, non-safety diesels, all the case.

But the equivalence is, is that they're inherently more safe from the standpoint of needing power, AC power. And so you have, you can either prevent it, deal with it or mitigate it through your inherent safety features. And that's the equivalency that we're looking at, at this point, is that correct?

MR. GILBERTSON: Yes. Yes, I think that's a fair way to characterize it.

MEMBER HALNON: I just want to get an example out there. It's not that we're not considering SBO it's that it's not as significant impact to the reactor core or wherever the fuel is at this moment.

MR. GILBERTSON: Right. And that's something that you would expect to come through the results of your PRA as you analyze the facility.

In addition to, you know, resolve step could inform what, you know, degree of defense-in-depth that you have. You can integrate the results of the PRA, the event sequences themselves. You can make characterizations about, you know, how much defense do I have or classes of types of responses to different

initiators and such. And then you can make decisions about that in your design, so yes.

MEMBER HALNON: Okay, thanks.

MEMBER ROBERTS: Yes, probably should let you move over to the next slide. This is Tom Roberts. But thinking about Bob and Dave's comments.

But the primary saying is, there's a carve out for DBAs, other than that there is like a frequency consequence thought process. Because it's, other than design basis accident events, because it's not really beyond design basis because it also includes AOOs and normal operation.

And so the principle is you want to have the lowest likelihood of consequence with the higher frequency events just a variation of that C-curve. And so it seems like, the way Dave explained it, the DBA is a carve out just to be able to, you know, I won't say check the box, but have a basis for the safety related equipment. But you still have the non-safety and special treatments for any equipment for the rest of the spectrum.

So is that the right thought process, that the DBA is the carve out but in reality it's the other than design basis that gets you the appropriate quality versus risk?

MR. GILBERTSON: I think that's generally a good way to frame that. Yes. That's in line with what I think I've been trying to kind of explain previously. Yes.

MEMBER ROBERTS: Because it seems like it's largely consistent with LMP, except that LMP is a lot more specific. It's got check points at ten to the minus four, keep it common with BDBE, and parts of ten to the minus seven you become a BDBE, whatever they call it.

And there are requirements for each of them. You got the cliff edge effects and the design, defense-in-depth assessments required for even beyond the five to ten minus seven. You got the BDBE requirements and the special treatment requirements for the BDBE range.

So it seems like you can accomplish that and the, you know, the licensing route can give me your standards. And you'd expect each application to have the equivalent of LMP in terms of these are the criteria that we expect to be versus frequency.

And, oh by the way, there are also DBAs that have a different -- okay, it sounds like I understand you now. So thank you.

One thing I noted, I did an electronic

search for the word cliff edge I the 800 page document, I didn't find it. So it seems like the concept is basically up to, we're going to have to address. But there is also, you know, frequency bound stated were implied in your next slide that defines what a, other than DBA is. So it sounds like basically, you know, the applicant can tell me what you think you need to be. And then you'll assess that on a case basis depending on what they tell you.

MR. GILBERTSON: Yes.

MEMBER ROBERTS: Okay, thanks.

MR. GILBERTSON: And then, yes. And then that goes directly to the flexibility that was intended in development as well.

MEMBER ROBERTS: You probably can move on.

MR. GILBERTSON: Okay. All right, we can move to, I think I finished with this slide. Go to Slide 16 please.

Okay. So moving on to the safety criteria for LBEs other than DBAs. Section proposed 53.220 would provide safety criteria for those. The identification and analysis of which would be required by 53.240 and 53.450(e).

In addition, the criteria, the safety criteria under 220 for LBEs other than DBAs, it would

consider a broader set of potential scenarios related to both internal and external hazards. Just like we've been talking about. It's a broader spectrum of risk consideration.

And the requirements under 53.220(a) would establish those connections between the capability and reliability of SSC design, human actions and programmatic controls for the wide spectrum of plant conditions considered. And these requirements would also explicitly address consideration of defense-in-depths, such as the balance consideration of event preventing and against, and mitigation of radiological releases.

So Paragraph 53.220(b) is the subject of the next SRM item, which is Item 2. And the safety criteria in 53.220(b) of the draft proposed rule was revised to remove the Commission's quantitative health objectives for QHOs and their related risk matrix of the individual early fatality risk and the individual latent cancer fatality risk.

So the requirements in the Part 53 proposed rule would include the use instead of a comprehensive risk metric, or set of metrics, and associated risk performance objectives against which the calculated values of the risk metric, or metrics,

would be compared.

So that's very -- That's similar to the paradigm discussed relative to the QHOs and those risk metrics.

So in that regard, an application to the Part 53 proposed rule would be required to include a description of the methodology for the use of the proposed comprehensive risk metrics and that would include, among other things, an explanation of the initial boundary conditions, initial conditions, and key assumptions that would be used to develop and calculate the risk metrics.

The comprehensive risk metrics or set of metrics and their associated risk performance objectives would support a performance-based approach to developing an appropriate combination of design features and programmatic controls to prevent or mitigate LBEs other than DBAs.

So that's another way of saying what we sort of already have been talking about, it's all working together, the analyses and form, how this, the DBAs are selected, and then, which, of course, informs your safety-related SSCs.

Okay, Slide 17, please.

MEMBER KIRCHNER: Anders, before you go on

this, this is Walt Kirchner again.

MR. GILBERTSON: Yes.

MEMBER KIRCHNER: Pragmatically, are you thinking that it's the frequency consequence curve or are you thinking that each reactor technology would have its own set of metrics and performance objectives?

MR. GILBERTSON: Well --

MEMBER KIRCHNER: What's the surrogate that you see here versus things like, like I would presume in the back of your mind is things like CDF and LERF, but those don't necessarily work for other technologies, so, you know, pragmatically how do you see this playing out since this takes the place of QHOs?

MR. GILBERTSON: Right. It could be the frequency consequence curve that is described in NEI 18-04 and surrogates could be developed as well. Like you mentioned, CDF and LERF is probably not going to work for most technologies, but something could be developed to that end.

That's certainly possible, but LMP is certainly going to be one way that you could meet the requirements. I will talk about that a little bit more actually on the next slide as it relates -- Yes?

MEMBER DIMITRIJEVIC: Hi, there. This is Vesna Dimitrijevic. I came into a problem here with the just naming conventions, comprehensive risk metrics because they could be interpreted as comprehensive set of metrics or comprehensive risk metrics which describe this overall risk.

So, you know, when I think about that I try to visualize what would that be. I mean what -- In your opinion how does this, you know, we were talking about the C-curve or the CDF and LERF, does a different risk metric capture the uncertain risk?

So in your opinion what does actually a comprehensive risk metric apply? Is this the comprehensive selection of the metrics or we are just talking about comprehensive risk?

MR. GILBERTSON: It would be the latter, it's comprehensive risk. So the idea is that it would represent the total risk of a facility of a design --

MEMBER DIMITRIJEVIC: Right.

MR. GILBERTSON: -- that it would be exposed to.

MEMBER DIMITRIJEVIC: Right, but when you say, comprehensive set of metrics, that could be interpreted both ways. That's just my point, you know.

MR. GILBERTSON: Yes.

MEMBER DIMITRIJEVIC: I mean it should be just comprehensive risk or metrics which models comprehensive risk, you know. So when you say comprehensive risk metrics it could be talking about a different selection of the, you know, metrics to model the risk.

MR. GILBERTSON: Yes. So --

MEMBER DIMITRIJEVIC: That's just my comment. I have a problem with terminology here.

MR. GILBERTSON: Sure. Okay, yeah, and I appreciate the comment. We do have a guidance document that is under development that eventually is going to move forward with the draft final rule package that goes to the comprehensive risk metrics, the risk performance objectives, how those would be evaluated by the Staff, et cetera, so --

MEMBER DIMITRIJEVIC: But I mean if you -- right, except for the overall risk metrics, I mean that would make sense and just --

MR. GILBERTSON: Right. Right, yeah. And, you know, we were taking language that borrowed language directly from the SRM to stay consistent with what the Commission has directed us to do.

So the further explanation of that will

come through in the guidance, okay.

MEMBER ROBERTS: But the term, comprehensive risk metrics, is supposed to be say capture, also say defense in depth considerations. I could imagine, you know, you consider dose, obviously, but maybe fuel temperatures and other metrics, you know, that would all have criteria.

Is that idea? Or when you mean comprehensive, it's not limited to, you know, a frequency consequence with dose, for instance, it goes beyond that to include other potential say engineering limits --

MR. GILBERTSON: Well I guess --

MEMBER ROBERTS: -- and likelihoods?

MR. GILBERTSON: Yeah, so it's really more focused on how those risk metrics are used to form the PRA and the results of that and how that is integrated into the design decisions.

There may be other design deterministic risk metrics or something for defense in depth that are used as part of the overall decision-making process, but in terms of the evaluation of the event sequences, the risk produced from that, whatever your output is from the PRA, that's really what those risk metrics are representing.

Now the PRA can be used in conjunction with, and we talk about this in the proposed rule, with other generally accepted approaches or systematically evaluating engineering systems.

So there could be a blending of the PRA together with other types of risk-informed analyses that may use different metrics and those are brought together to, again, use that in the overall design decision-making process, if that makes sense.

MEMBER ROBERTS: Yeah. No, I understand. I'm trying -- Yes.

MR. GILBERTSON: Okay. Okay, let's move on to Slide 17 in the interest of time here. So I'm going to try and cover a little more ground here so we can keep moving forward.

So the Section 53.450(e) is addressing, like it does for (f) in the DBAs, this is for the LBEs, other than design-basis accidents, how those would be analyzed and it provides requirements for analyzing the LBEs, the evaluation criteria for LBEs, a starting and end point for analysis of an LBE, and the process for identifying risk significant event sequences.

So, again, if you pair that, if you set that side-by-side with the LMP methodology you can see

the commonalities that come out of that, the licensing basis of the categories, for example, and how, you know, the risk significance of event sequences are determined.

So these requirements would address this analysis to demonstrate the performance criteria in 53.220 are satisfied, but also to show that the evaluation criteria to find for each LBE or category of LBEs would also be satisfied.

So, again, that's the sort of binning into these different categories of LBEs, the AOOs, unusual event and very unusual event sequences.

So the evaluation criteria for specific LBEs or categories of LBEs would be defined in terms of limits on the release of radionuclides for maintaining the integrity of one or more barriers used to limit the release of radionuclides and would reflect a graded approach of allowing lesser potential consequences for more frequency events.

Again, the LMP would be one way to do that and was used to inform the development of these requirements.

Slide 18, please. Okay, so talking a little bit more about the comprehensive risk metrics and the associated risk performance objectives.

As was directed in SRM Item 2, the preamble of the proposed rule was expanded to explain that comprehensive risk metrics should consist of a proposed plant risk metric or set of proposed risk metrics that approximate the total overall risk from the facility and that address the range of possible plant configurations and associated internal and external hazards to the extent practicable.

The risk performance objectives associated with the proposed comprehensive risk metric or set of metrics are pre-established acceptable values that are used to compare against measured values of risk metrics as part of risk-informed decision-making.

So as we have already talked about, one of the most relatable examples of this are the quantitative health objectives described in the Commission Safety, well Policy Statement, and the analogous comprehensive risk metrics would be i.e. of the individual early fatality risk and the individual latent cancer fatality risk.

So those could be used to form the basis for meeting the proposed requirements of 53.218(b), but, again, that could just be one way. There may be other proposals and, of course, those would have to be evaluated on a case-by-case basis.

So the use of the comprehensive risk metrics and associated risk performance objectives would provide a logical performance objective to support risk management approaches described in the various subparts of the Part 53 proposed rule and applicants could choose to propose and seek NRC approval of comprehensive risk metrics and their risk performance objectives, including the use of surrogate metrics as we previously discussed.

At the moment we don't, we haven't seen any of any surrogates per se for non-LWRs, but they could be developed and proposed.

Again, I will just emphasize that the Staff are developing guidance on determining the acceptability of proposed comprehensive risk metrics and the risk performance objectives and that is expected to go with the draft final rulemaking package.

Slide 19, please. Okay, so the safety functions are addressed under Section 53.230 and this section would specify that limiting the release of radioactive materials from the facility is the primary safety function and would need to be maintained over the life of the facility.

The primary performance metric used

throughout the Part 53 proposed rule would, therefore, be limiting potential offsite consequences, so, for example, dose to a hypothetical individual.

53.230 would also require identification of additional or subsidiary safety functions that are needed to limit the release of radionuclides, which could include controlling processes related to reactivity, heat generation, heat removal, and chemical interactions.

So the primary and these additional safety functions would be required to meet the safety, to satisfy the safety criteria under 53.210 and 220 if an assumed LBE were to occur at a facility and would be fulfilled by the design features, human actions, and programmatic controls that are addressed throughout the Part 53 proposed rule.

So, again, as we have mentioned previously, this proposed rule would provide flexibility to applicants and licensees in identifying, implementing, and maintaining the safety functions supporting retention of radionuclides for facilities of varying sizes and new technologies.

MEMBER HALNON: So, Anders, this is Greg. The functional containment comes to mind. Have the Staff counted on a succinct definition of that and we

understand that completely what that's going to look like, it seems like there was some fuzziness earlier on about definitions and what it actually looks like from a nuclear perspective; have we settled on what that is because it certainly falls into this first bullet?

MR. GILBERTSON: Right. Well there is, of course, the SECY paper I am not able to, I don't recall the number off the top of my head, but that specifically address functional containment.

Bill Reckley is the author of that paper and that lays out the concept of how we think about it and, of course, I think we're seeing what that looks like with applications that are coming in and we're having to assess that in that context.

MEMBER HALNON: Have you been awaiting feedback from the Commission on that paper?

MR. GILBERTSON: That was --

MEMBER HALNON: That was awhile back?

MR. GILBERTSON: Yes, that's a little while back.

MEMBER HALNON: Okay.

MR. GILBERTSON: So that was --

MEMBER HALNON: That's what we're still operating off of then, right?

MR. GILBERTSON: Yes. Yes.

MEMBER HALNON: Okay. I remember what you're talking about. Thanks.

MR. GILBERTSON: Yeah. Okay, we can move on to Slide 20, please. Okay, so 53.240 would require applicants to identify and analyze the LBEs for the purpose of demonstrating that the safety requirements in subpart B have been satisfied.

One or more of which of those LBEs must be a DBA. The LBEs, as shown in the definition down here, are those unplanned events that would fall into one of the four categories and that includes anticipated event sequences, unlikely event sequences, very unlikely event sequences, and the design-basis accidents.

So all of those were the four categories, but we have been discussing how the DBAs serve a different function than the first three mentioned there.

The analysis of the LBEs under 53.450 would help ensure that the related estimates of offsite consequences would be below the safety criteria identified under 53.210 and 53.220 and that the SSCs, personnel, and programs address the safety functions identified in the proposed 53.230.

So this would reflect the historical and continued importance of evaluating unplanned events as part of the licensing of commercial nuclear plants.

And, finally, 53.240 would also require that the analysis of LBEs confirm the adequacy of design features and programmatic controls in meeting the safety criteria under 53.210 and 220 and that they would be used to establish related functional requirements for SSCs, personnel, and programs.

Okay, Slide 21, please. So getting to defense in depth, this section, 53.250, would establish requirements for defense in depth and those are based on longstanding philosophy of providing defense in depth to address uncertainties about the design, operation, and performance of the commercial nuclear plants.

As we have discussed previously this would provide flexibility in how applicants would propose to demonstrate compliance with the high-level safety criteria as would be informed by defense in depth and these requirements would include that no single engineered design feature, human action, or programmatic control, no matter how robust it is, should be exclusively relied upon to address LBEs other than DBAs.

So the phrase, engineered design feature, would not preclude crediting inherent characteristics within the design and analysis, so, for example, inherent characteristics of fuel performance, as an example.

MEMBER ROBERTS: Anders, this is Tom Roberts. That last bullet implies that you can rely on a single engineered design feature for a DBA. Is that the intent?

MR. GILBERTSON: No. There would still need to be -- you wouldn't be able to do that and meet the defense in depth requirement as it's proposed here. So there still needs to be a demonstrated balance between prevention and mitigation. So yeah. Does that help?

MEMBER ROBERTS: I was thinking that the words other than DBA don't really need to be in that bullet. And I guess I'm wondering in general about the definition, that a DBA is not really a DBE, and a DBE comes out of a PRA that these are the event sequences that could happen. A DBA is kind of a deterministic combination to bound several of those for the purpose of SSC determination, just -- so it just seems like the concepts are different, LBE and DBA.

And maybe just something to think about is to take out that word group other than DBAs. Just make it DBA requirements and LBE requirements. Just a thought.

MR. GILBERTSON: Okay. All right. Appreciate that --

MEMBER HALNON: No, it's a good thought. And I was looking at you thinking that you would probably pounce on this one. The same point I was making earlier, you know, there are two different things, at least, we can try to tackle with your analysis. And one, of course, is the deterministic evaluation of the performance of safety-related equipment, and then there's the other that's the more holistic one that looks at the subject of defense in depth.

So yeah. I'm with you, Tom, that the other -- it just doesn't belong here because it -- you know, of what we're talking about. The other reason I was looking over -- I thought you might pounce on cliff edge effects because this is where I think you would put it -- you know, you put mention of it as far as the search.

Oftentimes, I think applicants have a difficult time with, what does it mean? What is the

defense in depth requirement? And they're really looking for the actions that are expected of them for that purpose. Now, granted, a rule is intended to be higher level. You capture these things in guidance. And we're doing better, certainly, seeing that a number of examples of -- at least in regulatory guidance where cliff edge effects comes in.

However, the rule is what kicks it all off. And certainly, I would feel like this is a good place at least to mention that there's an expectation as part of defense in depth -- is there is this comprehensive search for cliff edge effects. And this is Engineering 101 where you create the box and you protect the boundaries of the box. Cliff edge effects are the boundaries.

MEMBER ROBERTS: And the reason I didn't raise that is because I read the definition, very unlikely, in the draft language. And it would encompass basically any frequency. So it's up to the applicant to figure out how to bound that, and that would, I think, require a search for cliff edge effects and the assessment of defense in depth down to some sort of lower cutoff.

And, oh, the question of what the lower cutoff is is yet another question we've discussed in

other meetings, and still waiting for the definition of what that might be. But yeah, I think the language here, Greg, would cover it, but it puts a lot of onus on the applicant and the engagement with the regulator to figure out where that bound is.

MR. GILBERTSON: Okay. Thank you. And I guess, as far as a DBA being able to rely on a single function or piece of equipment, put in the context of the LMP methodology, the DBAs are derived from the DBEs. So, if you impose the requirement on your DBEs, there should be -- that should transfer into your DBAs. You would necessarily not be able to have that. You'd have some set of safety-related equipment. But the point that you made about the language -- that's a fair point. I'll definitely take that back.

Okay. Let's move on to slide 22. Okay. So the next couple of slides are going to focus on changes made in the Part 53 proposed rule to address Item 4 from the SRM. This is -- relates to the concept of as low as reasonably achievable, or ALARA, and directed the staff to retain the use of design objectives to demonstrate how effluent released would be limited, consistent with 10 CFR 50.34(a), which provides objectives related to control of radioactive effluence.

So let's move to slide 23, please.

MEMBER ROBERTS: Real quick, Anders.

MR. GILBERTSON: Yes.

MEMBER ROBERTS: The definition of commercial nuclear plant -- is that going to be reiterated from, I guess, the recent NTAC patent definition of labor -- or maybe it was the one before that -- relative to the -- how much revenue goes into operating the plant, that sort of thing? Is that going to be reflected in the regulation, or are we going to be arguing what's commercial versus not when you're actually selling electricity and you're not quite commercial?

MR. GILBERTSON: Right. I guess what I would say at this point is that we are working closely with the team that is addressing implementation of the ADVANCE Act. Can't specifically say, as far as Part 53 rule is concerned, how we would necessarily implement that. But there's close coordination. Perhaps there will be comments on that that we'll receive --

MEMBER ROBERTS: I'm interested in that only because, being on the other side for most of my life, the safety culture aspect of business objectives versus commercial -- I mean safety objectives -- it's

always -- I'm not going to say a struggle, but there sometimes can be -- and I'm interested in how we're going to parse that out in the actual reviews of these applications.

MR. GILBERTSON: Okay.

Okay. Let's move on to slide 23, please.

Okay. So this discussion, the SRM item related to ALARA -- it cuts across a number of different requirements. So we're kind of presenting all of them here -- illustrate, talk a little bit about how we address that.

53.260 and 270 were revised to more directly reference to radiation protection requirements for normal operations under 10 CFR Part 20. Section 53.25 was likewise revised to emphasize that the design should support meeting the requirements for radiation protection, the Radiation Protection Program, and it also includes a footnote related to the use of the 10 millirem per year design objective, which serves the purpose of Appendix I to Part 50 under the current requirements.

So this means that if an applicant can show that the proposed design features will support meeting the design objective, there should be no need to look for additional improvements and terms of

further reducing public doses from normal operation.

For Section 53.430, this was similarly revised as it relates to worker protections. In addressing the parallel to 10 CFR 50.34(a), there would need to be an analysis of expected releases and doses to the public. And this would address inventories, or address the anticipated inventories, their locations, controls thereof, to show that the expected doses would meet the Part 20 requirements.

Section 53.50 on the Radiation Protection Program was also revised and includes requirements that are equivalent to environmental technical specifications such as an off-site dose calculation manual, because Part 53 does not include a proposed requirement for environmental specifications.

Requirements on annual reporting under 53.1645 were also revised to address this item but are largely consistent with current requirements. And the next set of requirements, going to the end of --

MEMBER ROBERTS: Stop you on 1645.

MR. GILBERTSON: Oh. Sure.

MEMBER ROBERTS: So I asked you earlier about the role of ALARA. And what it looked like in the prior proposed language was ALARA working its way more in design requirement space as opposed to

operation space. Now a lot of that's been cut out. But this section still says design requirements of ALARA. And I don't know if that was an oversight. That doesn't seem to jibe with everything else that has been edited out.

But it still says ALARA and design requirements in the same sentence, and it doesn't seem consistent with the other edits that have been performed. And I don't know if that was intentional, unintentional, but it might be worth circling back on 1645.

MR. GILBERTSON: Okay. Okay. Thank you.

So, getting to the last portion of this table here, this all relates to the different licensing applications, and it becomes a little more nuanced here when you're talking about differences between pre-operational and operational stages with licensing.

So, for pre-operational licensing stages, most of the revisions to the draft rule were effected to the design certification requirements because many of those requirements are references in other portions of the licensing requirements.

So, for design certification or construction permit, the applicant describes the

feature, the design features, but does not need to provide the functional design criteria. In addition, an applicant would describe how programmatic controls, including monitoring programs, would support meeting the safety criteria and would be used to supplement a given design feature.

So, for example, this would mean describing that a filter, which would be the design feature, would be used versus talking about how efficient that filter necessarily needs to be, which would be more the functional design criteria. And similar revisions were made relative to requirements for occupational exposures. So, again, that's in the context of pre-operational stages, so construction permit, design certification, other manufacturing license, SDA, that sort of thing.

For licensing phases related to operational stages, the applicant would need to provide information about the design and the Radiation Protection Program. So, in that way, the Radiation Protection Program would address the functional design criteria and how everything comes together to achieve the safety function and meet the rule.

So, for example, a filter is used as the design feature. It must perform within certain

specifications, and the performance of the filter is monitored via programmatic controls. And all of that would be described as part of that -- meeting those requirements. And again, similar changes were made relative to the requirements for occupational exposures.

Okay. Slide 24, please.

MEMBER HALNON: So, Anders --

MR. GILBERTSON: Yes.

MEMBER HALNON: -- we had it down on the agenda a 10:15 break.

Dave, I think this is an appropriate place to take ten minutes, if you agree.

CHAIRMAN PETTI: Yeah. Sounds good.

MEMBER HALNON: Okay. So we'll be in recess until 10:25.

(Whereupon, the above-entitled matter went off the record at 10:15 a.m. and resumed at 10:25 a.m.)

CHAIRMAN PETTI: Are we ready to start again?

MEMBER HALNON: Yes, Dave. We just got ready to go, so. So you can go ahead and get started. Anders, go ahead.

MR. GILBERTSON: Okay. So we'll get

started again here. We're on slide 24, and this is getting into subpart C, which is addressing design and analysis requirements. Go ahead and move on to slide 25.

Okay. So this diagram is really just intended to be kind of a helpful tool to put the overall structure of the Part 53 requirements into a general kind of framework. And it really kind of illustrates the systems engineering approach that was taken to develop the structure of the proposed rule. I'll try and go through this relatively quickly because I'll emphasize it as I get into the sections under subpart C.

But the top level -- or the top chevron there -- we've already talked about safety criteria on 210 and 220 and that the subsequent layers underneath that safety functions and so forth are used to demonstrate that those will be met.

The safety functions are necessary to ensure that the safety criteria are met. So, again, it's just asking, what are the functions that need to be present to make sure that the radiological releases are limited? So, for example, that's controlling cooling, heat production, containing radionuclides.

The next are the design features which are

kind of requirements here in subpart C, and those are used to fulfill the safety functions that have been identified. So, for example, that would be whatever the actual system is -- do control cooling, which would be achieved with pumps, valves, et cetera -- heat exchangers.

And then at the bottom level would be the functional design criteria, which we're going to describe how this design feature is looking to perform to satisfy their design function. So, for example, a cooling system, functional design criteria would talk about things like minimum flow rate of a pump, heat exchanger capacity, et cetera.

Okay. Next slide, slide 26.

So what does this look like in subpart C? Well, we've got Section 53.400, which requires design features be provided to satisfy the safety criteria. Section 53.410 would require that functional design criteria be defined for design features relied upon to demonstrate that the consequences of DBAs are below the 53.210 safety criteria and of course will be analyzed per 53.450(f).

53.415 would require that safety-related SSCs be protected against or designed to withstand the effects of natural phenomena, human-constructed

hazards, such that the safety-related SSCs remain capable of performing their associated safety functions under those conditions and up to the magnitude of the design basis external hazard levels as would be identified and characterized under 53.510, which I'll get to in a few slides.

MEMBER MARTIN: I'll use this as my opportunity to mention the word hazards analysis. And this is not just Part 53, but obviously, the SSCs have to be protected against all hazards. We do elevate external hazards in particular, for good reason.

But I do think the emphasis and the potential de-emphasis of internal process hazards is confusing, if not unproductive, because of course they do. I mean, hazards -- you know, anything that's going to disrupt the operation of the power plant are a concern, a concern to everyone.

And I just feel like when I went through it, again, doing the same thing that Tom did, searching for my favorite word, hazard analysis, other than the sections with fire hazard -- which have been cut out, and I still haven't quite figured out where that went -- it doesn't show up other than external hazards in particular.

There is one location, internal and

external, but I'm kind of sensitive to what I perceive is de-emphasis on a holistic view of hazards. And of course, part of that holistic view is the necessity for hazards analysis. And that gets in the guidance, and the guidance will talk about that.

But the main point here is not to overemphasize a particular hazard at the potential risk of de-emphasizing a broader -- and just like there's a search for cliff edge, prior to that, there is a search for hazards. And it is agnostic to what causes the hazards.

And I just feel like what's been written in there is -- you know, seems to put all the emphasis on external hazards, when ultimately, hazards come in different forms.

(Off-microphone comments.)

MR. GILBERTSON: All right. Moving here -- so the requirements, Part 53.415 -- those would support the use of traditional deterministic approaches or probabilistic approaches for determining and protecting against external hazards, including probabilistic approaches under development for seismic and some other external hazards.

I guess I might just add that while the rule language does emphasize external hazards, the

foundation of the PRA is the plant -- or the facility design performance as it relates to the internal events model, which is fundamentally the individual equipment failures. Other types of hazards are sort of overlaid to that and then brought forward. But --

MEMBER MARTIN: I mean, as a historical basis, if you go back 60-plus years ago, we had kind of a technology-inclusive approach, right? Because the technology -- whether it was light-water reactors or sodium reactors, et cetera -- we hadn't quite figured out what that was going to be.

Rather than a safety analysis report, we headed a hazard evaluation organization at the AEC. And that terminology, I think, was because -- you know, when you don't know what the technology is that might be proposed, it really comes about what are the hazards -- the emphasis is what can hurt people and what can cause that.

And personally, I have a bias towards the term hazards evaluation, but it is in the DNA of the regulator, whether it's NRC or the AEC, to use that terminology -- begin with that terminology -- and everything flows from hazards analysis.

Obviously, all these things that we're talking about will show up when you do it. And you

attack all those things in different ways. And yes, we have reg guides that are already attacking all the external hazards because, of course, it's been emphasized in Part 50 and Part 52 already.

But there isn't a overarching view other than you should do something. Now, more likely, that just gets addressed in guidance, but I do feel like it should be emphasized at that highest level of hazards analysis without putting a label of external, necessarily, in front or in part. We just -- you begin at a higher level, and then the hierarchy flows from there.

MEMBER HALNON: Walt, you had a question?

MEMBER MARTIN: Yeah. I'm a little preachy, but --

MEMBER HALNON: I didn't mean to cut you off, but Walt had a question.

Go ahead, Walt. Walter, you're muted.

MEMBER KIRCHNER: Thanks. I didn't hear you, Greg.

I'm curious about the choice of wording here on this sub-bullet for 53.415. Just for context, the existing fleet -- many of the systems that are relied on in a holistic sense for the overall safety of the plant are protected by a hardened containment.

So -- and specifically, those that are safety-related are also protected accordingly.

Why did you use safety-related -- this sounds restrictive because my concern is this, that if an applicant has a very limited number of safety-related systems that are protected against external hazards or has a very limited number of safety-related systems, period, the plant as a whole may not be adequately protected against external hazards.

The general design criteria uses the terminology, important to safety. So SSCs important to safety should be -- shall be designed and protected. So do you really mean this, just safety-related, those that come out of the SSC classification process -- those will be protected against external hazards, but the rest of the systems may not? I find this a concern.

MR. GILBERTSON: Yeah. Yes, that is what we mean. And I think that when that notion is taken together with the things that would be done to satisfy the requirements for defense in depth and understanding that if on a plant-wide basis, you have a limited set of or a smaller set of safety-related SSCs that are protected against these external hazards, and then a much larger set that are not

necessarily or to a much lesser degree, that's the type of thing that the PRA would hopefully reveal, that okay, your layers of defense for the event sequences that extend into the very unlikely event sequence frequency range -- whatever that would be defined as -- you have less capability there because you're not protecting the systems against --

MEMBER KIRCHNER: Well, but wait a minute. Stop. I mean, there are a lot of external events that are highly likely. Tornadoes are a good example, protecting against telephone poles and other tornado-induced missile hazards. That's a very real, not highly unlikely, event -- flooding, et cetera.

I have a general concern with this because, especially as we go to smaller reactors, these are not likely to have the kind of structure that we see with the current fleet and the inherent protection that comes from something like a large containment structure.

So external hazards may be one of the Achilles heels for some of these smaller microreactors and other systems that are under consideration. And so, to make this restrictive, if they were to use 53 -- I'm just concerned with the restrictive nature of this.

Now, yes, you're correct, a thorough PRA would look at those other natural hazards, whether it's a natural phenomena, a fire, and/or -- you know, I'm looking at criteria 2, 3, and 4 of the GDCs. Those may be the dominant hazards for these smaller, lightly deployed systems. So to just restrict it to safety-related just seems to be not the holistic look that one would expect.

And so, yes, you're right, if it's thoroughly exercised, the PRA would look at that and then tell you additional measures may be necessary.

MR. RECKLEY: Yeah. Anders, this is Bill, if I can -- Bill Reckley, if I can chime in here a little bit so people don't misinterpret what we're saying here.

415 is saying safety-related SSCs have to be protected at least up to the design basis external hazard level, which would be the traditional approach for safety-related equipment and defining a design basis external hazard level, like a safe shutdown earthquake.

However, non-safety-related but safety-significant SSCs will be evaluated against seismic events, as Anders said, through the PRA assessments, and that could include external hazards that exceed

the design basis external hazard level -- an earthquake that's stronger, for example, higher magnitude. And the fragility of those SSCs will get looked at and considered within that PRA. And additional special treatments may be assigned to those if they have to be in order to meet those criteria.

So I just don't want to leave the impression that only safety-related SSCs are being protected. No. They're being protected against a minimum. All SSCs that are safety-significant or, using the old term, important to safety will get looked at from the effect of external hazards.

(Simultaneous speaking.)

MEMBER KIRCHNER: Okay. Thanks, Bill. Maybe I just read too much into this safety-related lead-up to that. But yeah. If, indeed, you feel that the rule as you've got it, as written, is encompassing, then I'm okay. I was just reacting, maybe, to the graph presentation.

MR. RECKLEY: Right. And I'll say rule as we intended -- comments and other things will say whether we actually captured it correctly.

But sorry, Anders. Go ahead.

MEMBER KIRCHNER: Yeah. Thank you.

MR. GILBERTSON: Okay. Thanks, Bill.

Okay. So I'm going to keep moving here. The proposed 53.420 -- so this just relates to the functional design criteria for design features that play a significant role in demonstrating that the safety criteria in 53.220 would be satisfied. So, again, you see this dichotomy between 410 and 420, between the design basis -- licensing basis events for other than design basis accidents.

As Bill was kind of alluding to here, the SSCs that are determined to be safety-significant would have associated requirements for special treatments. Those would be provided for under 53.460. We don't have a slide on that later on, but I'll just generally say that in special treatments, it would generally refer to those measures taken beyond the procurement and installation of commercial-grade products to provide confidence that the SSCs would comply with the applicable functional design criteria.

Okay. Let's move to slide 27, please.

Okay. So this is now getting into the design requirements. The proposed 53.440 would provide various requirements specifically included to ensure the design features required by 53.400 would comply with the functional design criteria required by 53.410 and 420. So the requirements in 53.440 would

be met through design practices, consideration of testing and operating experience, and various assessments of LBEs and other potential challenges.

I think I'm going to emphasize here that 53.440(a) had an item that was added back to it that related to SRM Item 7, and this directed the staff to include the requirement for a design experience program that corresponds to 10 CFR 50.34(f)(3)(I), which is under the Three Mile Island-related requirements. And we did that. That was a relatively straightforward change.

Otherwise, 440(a) would provide requirements to demonstrate that each of the design features would -- well, I'm sorry. This -- it closely aligns with the requirements for 10 CFR 50.43(e) regarding the use of analyses, test programs, prototype testing, and operating experience to demonstrate the performance of a given design feature. Just wanted to note that.

The rest of these -- I'm going to go through these relatively quickly so we can start to catch up a little bit. 440(b) would require that the design features be designed using generally accepted consensus codes and standards that would have been endorsed or otherwise found acceptable by the NRC.

53.440(c) would require materials used for the safety-related and non-safety-related but safety-significant SSCs be qualified for their service conditions over the design life of the SSC.

53.440(d) would require consideration of possible degradation mechanisms for materials and equipment to inform both the design processes and development of integrity assessment programs. And this would be active during the active operational phase in accordance with those requirements in subpart F of the proposed Part 53.

Sections 53.440(e) and (f) would provide design requirements similar to the existing requirements in Parts 50, 52, and 73 for protections against fires and explosions and consideration of safety and security together in the design process.

53.440(g) and (h) would require that commercial nuclear reactors have the capability to achieve and maintain subcriticality and long-term cooling, which includes potential for further actions to completely shut down and service a facility that has already achieved a safe and stable end state.

53.440(i) would require consideration of the number of reactor units and other significant inventories of reactor materials riveting to the risks

to the facility and public health and safety.

53.440(j) would provide requirements similar to those in 10 CFR 50.150 as it relates to possible impacts of large commercial aircraft.

MEMBER MARTIN: I'll just stop you on that one. I find this one particularly interesting because it effectively puts a deterministic design requirement on plants that are supposed to be, you know, lower profile, smaller. It doesn't seem to have the risk-informed element to it.

And I wonder if this just forces everyone to put their plant underground, you know, or firmed up or -- we don't want hardened containments, right? I don't know how much feedback you all have gotten on this particular one, but I don't know what the right answer is. I'm not going to judge it, per se, but I do think it creates a design requirement that it's going to be more expensive, for sure, if you want to stay above ground, for instance.

I don't know. I mean, have you gotten feedback particularly on this requirement, and is there pushback? Or what's the conversation from your engagement with industry folks, specifically on aircraft impact?

MR. GILBERTSON: I don't think that we

have gotten any specific feedback, but I know that other staff have been working specifically on this and some other related stuff. I guess maybe I would ask Nan or Bill if they have any additional insight to share.

MS. VALLIERE: Yes. This is Nan Valliere again from the Advanced Reactor Policy Branch of NRR. So Anders is right that we did not get a lot of feedback on this particular item from external stakeholders during development of the draft proposed rule.

However, since that time and with all the activities related to microreactors that have been undertaken, this area is an area of significant discussion. So, yes, we are looking at what other possibilities could be undertaken with regard to aircraft impact requirements for microreactors. Of course, the aircraft impact rules are directed by the Commission. So it was -- we couldn't exactly leave it out completely, but we are -- yeah, the staff is looking at ways --

MEMBER MARTIN: Yeah, it's just more risk-informed elements to it. And I don't -- no one knows the answer today. Otherwise, I wouldn't have asked my question. But nonetheless, it's good to hear that you

are getting some feedback now and that you're synthesizing and processing it. So that answers my question.

MR. GILBERTSON: Okay. Thanks, Nan.

Okay. Moving along here, 53.440(k) -- this would require risk to public health and potential chemicals hazards of licensed material -- it would prohibit the diversity of reactor technologies designs that might be licensed. So this would be similar to the existing requirements in Part 70 that address both potential radiological and chemical hazards for licensed materials at fuel cycles.

53.440(l) would require that measures be taken during the design of commercial nuclear plants to minimize contamination of the facility and the environment, facilitate eventual decommissioning, and minimize generation of radioactive waste in accordance with 10 CFR 20.1406.

MEMBER BALLINGER: This Ron Ballinger. With regard to K, that's kind of an iceberg. You say licensed material. For some of these plants, they're going to use chemicals that may not be licensed, but they're there. And so those chemicals can create a hazard which is maybe worse than the licensed material. So does this account for that?

MR. GILBERTSON: This -- I would have to -- I think I'm going to have to get back to you.

Yeah, Nan?

MS. VALLIERE: So, again, this is chemical hazards associated with a licensed radioactive material and any chemical hazards that would be --

MEMBER BALLINGER: Okay.

MS. VALLIERE: -- intermingled with that. It's not talking about regulating chemicals by themselves.

MR. GILBERTSON: Yes.

MS. VALLIERE: This is consistent with how this is handled in other parts of the NRC regulations, division of duties between federal agencies.

MR. GILBERTSON: Yeah. Okay. Okay. So, moving on to 53.440(m), this would include a requirement equivalent to 10 CFR 50.68, providing options to either have criticality monitoring capabilities, meaning the requirements under 10 CFR 70.24, or to have restrictions on handling and storage of special nuclear material that would prevent inadvertent criticality events.

And finally, 53.440(n) would require that the design of a facility would need to reflect state-of-the-art human factors principles for safe and

reliable performance in all the settings that human activities are credited.

Okay. Let's move on to slide 28, please.

So Items (e) and (f) I've already touched on, so I'm going to talk about just some of the other items here. 450 -- 53.450 -- these would establish the requirements for analysis and would center on the use of the PRA in combination with the other generally accepted approaches for systematically evaluating engineered systems.

As we talked about before, the PRA is a key component in the proposed analysis requirements and reflects decades of improvements in PRA methodologies and the increase in use of PRA techniques in design licensing and oversight.

The Part 53 proposed rule would maintain a role for the NRC's traditional deterministic approaches, particularly, as I mentioned before, as it relates to DBAs and the defense in depth philosophy by including proposed requirements previously mentioned.

Specifically, 53.450(a) would provide a requirement for the use of a PRA, identifying potential failures, susceptibility to internal/external hazards, and other contributing factors to event sequences that can challenge the

safety functions and would otherwise support demonstrating that the safety criteria for 53.220 are met, or restricted alternative criteria essentially adopted under 53.470.

53.450(b) includes requirements for the specific uses of the PRA or specific uses of the analyses, which would include using the PRA together with those other acceptable techniques. And this would relate to identifying and categorizing LBEs, classifying SSCs, and evaluating defense in depth. So, again, it parallels to LMP per our pre, here or there.

And the increased role for the PRA necessarily means it would need to be developed, performed, and maintained in accordance with NRC approved standards and practices. And to that point, 53.450(c) would require periodic maintenance and upgrading of the PRA, which would ensure that there's alignment between the supporting analyses and the design and performance of plant equipment, programs, and procedures and other factors associated with meeting the 53.220 criteria.

These periodic assessments -- they would be performed by licensees but would be supported and complemented by NRC inspections and programs to assess

new or revised information for topics, such as chemical hazards, operating experience, et cetera.

53.450(d) would require that computer codes used to model the plant response and behavior of the barriers to release of radionuclides -- that those would all be qualified for the range of conditions that are being simulated across the range of unplanned events.

I'm going to skip now down to 53.450(g), and this would require that the analyses are performed to support the design requirements of 53.450(e) on fire protection and 53.440(j) on aircraft impact assessment, and the 53.425 requirements on using design features and programs to control doses to members of the public resulting from normal operation.

MEMBER MARTIN: The statement there -- doses to members of public -- how is that different than E and F? I mean, ultimately doses is a principal thing you look at. Is that something unique in other required analysis associated with doses?

MR. GILBERTSON: I guess it gets just to ensure that you're meeting the requirements for 10 CFR Part 20 as it relates to normal operations, is essentially what that's relating to.

MEMBER MARTIN: Okay.

MR. GILBERTSON: So your analysis was part of -- analyses were supporting that and demonstrating this.

And so I'll just note here, because we're -- we'll move on now to the next subparts --

DR. BLEY: Anders, this --

MR. GILBERTSON: Yes.

DR. BLEY: This is Dennis Bley. I hadn't thought about this in the past because we haven't reviewed an application for -- I'm thinking microreactors now. But in G, you flagged a few things that are typically a measure of importance in the analysis for our large LWRs.

But with these sometimes portable systems, there's probably different hazards. They could get up close to the chemical facility that might have a BLEVE. That's a different kind of fire protection where they could be hit by other things than aircraft that might be very significant, and I think we're kind of locked here on what we know from LWRs.

And I haven't searched that in my own mind, but I wonder if you folks have. And I think you ought to.

MR. GILBERTSON: Okay. I would just maybe say that, you know, other types of hazards for those

types of situations, they might be addressed through, like, industrial transportation hazards that a smaller reactor might be exposed to. Consideration of those types of scenarios, just as a thought.

MEMBER HALNON: Hey, Dennis, this is Greg. If you remember back to the reg guide that goes with the new Part 50.160, which I believe this is going to point back to, has you look at those types of hazards for the emergency plan.

DR. BLEY: Yeah, and I don't remember the details, if it's been really thought out. I was even thinking of -- you know, we know the military is looking to use microreactors. They might be impacted. We don't usually think about weapons, but they might be supporting a facility, and a testing of weapons nearby could end up causing an impact, too.

So, if that covers it well, that's great, Greg. But I don't remember the detail in that.

MR. RECKLEY: I'll go back and look to make sure on the slides, but I'm pretty sure that that was part of the development of the emergency plan, which would be this portion of the regulation as well.

MR. GILBERTSON: Okay. I'll just wrap up subpart C by saying I talked a little bit about the requirements for safety categorization and special

treatments. Those would be under 53.460. The requirements under 53.470 related to maintaining analytical safety margins used to justify operational flexibilities. That's using more restrictive criteria. That as well as 53.480 on earthquake engineering, there were no substantive changes to those. But those provide requirements to help inform those analyses that would -- just since we didn't include those items on the slides here.

Let's go to slide 29 so I can talk about PRA acceptability and then go through subpart D so I can hand it off to Bill here.

So the item -- SRM Item 3, this directs the staff, in part, to revise the proposed rule or preamble as appropriate to convey that consensus PRA standards should not be applied as a strict checklist of requirements for PRA acceptability determinations but should instead allow appropriate flexibility, considering how PRA insights are used together with other factors.

So the staff addressed this by revising the preamble to address the Commission's direction and provide additional explanation of how the NRC's regulatory guidance on PRA acceptability is currently used in decision-making and could be applied under the

Part 53 proposed rule.

And just to point out a few aspects of this, there are existing regulatory guidance documents -- Reg Guide 1.200 for light-water reactors and Reg Guide 1.247 for non-light-water reactors -- that provide guidance on acceptability of PRAs used in risk-informed decision-making. Those are available. They endorse light-water reactor and non-light-water reactor PRA standards.

But the use of those standards and the guidance is not regulatory requirement. And applicants under Part 53 may not need to follow every aspect of an applicable consensus PRA standard endorsed by the NRC. And to that point, I'll specifically talk about the non-LWR PRA standard. It includes a process for defining the scope and capability of a PRA supporting an application. And that's as based on the needs of the application.

So this is intended to -- you know, offers flexibility in determining the degree to which the PRA needs to be developed. It may be informed by factors such as design complexity and the degree of realism. So we touch on that in the preamble and note that those processes are available and that NRC determinations of PRA acceptability would include

consideration of the appropriateness of the scope and capability of the PRA defined by the applicant.

So that's essentially how we had addressed that item. There were no changes to the rule text itself.

MEMBER MARTIN: So basically, saying that, you're standing by your existing guidance on PRA acceptability.

MR. GILBERTSON: Yes, and that available processes can be used to define what that scope is. One could reduce the scope of the PRA and use that in conjunction with other types of analyses, like I talked about before, to create a sort of blended explanation of how those different risk metrics -- or whatever the outputs, how those work together to justify the safety case.

MEMBER MARTIN: Thanks.

MR. GILBERTSON: Okay. Let's move to slide 30, please.

Okay. So subpart D, it has the requirements for siting. And just to touch on these at a very high level, 53.500 would establish requirements for licensees and applicants to assess the impacts that a site and its environs may have on a commercial nuclear plant and potential adverse

health and safety impacts a plant may have on nearby populations relative to the characteristics of a given site.

53.510 relates to external hazards, and this is a callback to the reference and define basis external hazard levels; these have to be identified and characterized based on site-specific assessments of natural and human-constructed hazards with the potential to affect plant functions and with a focus on requirements related to seismic siting factors.

Section 53.520 would require applicants to identify and assess site characteristics related to topics such as meteorology, geology, hydrology, and other areas in a design and analysis required under subpart C.

53.530 would provide requirements for population-related considerations and maintain requirements and definitions similar to those used in Part 100 for an exclusion area, low-population zone, and population center distance.

And just to note, the NRC's longstanding preference for siting reactors in areas of low-population density would be maintained in Part 53 by using the current language from Part 100 for 53.530(c).

And finally, the proposed 53.540 would require that site characteristics be appropriately considered with other activities, such as the design and analysis performed and the (audio interference) under proposed subpart F.

Okay. So that concludes my portion of the presentation. Unless there are any questions, I would -- I'll hand it off to Bill Reckley to take the next of the slides.

MR. RECKLEY: Okay. Thank you, Anders. Thank you, Anders.

This is Bill Reckley, and I'll be covering subpart E.

If we go to the next slide, subpart E addresses construction and manufacturing. And if we can just go to the next slide on manufacturing, the higher level requirements on organization, management, and control for manufacturing under proposed 53.620 is similar to that defined for construction under 610. But the area highlighted here, fuel loading, is one we wanted to focus on today.

I would just note at the bottom there -- when we came to the ACRS in late 2022, we had a provision for fuel loading in the paper that you all looked at. In between your review and sending up the

draft proposed rule, the staff removed that. And now, basically, the Commission has instructed us to put it back, or put something back in its place. So that's what we're going to focus on in this section. Most of the other areas in 620 and 610 for construction were not changed.

So, got a question?

DR. BLEY: Yeah, Bill, Dennis Bley. I think in the introduction, it was pointed out that you folks were working on a white paper related to manufacturing. Is that related to this or is it related to something that might be changing in the draft rule? What's it's --

(Simultaneous speaking.)

MR. RECKLEY: Yeah, I'll get to that in a -- I'll get to that in a slide or two. But shortly, what that relates to is that the SRM also directed us to explore the potential to allow operational testing within the manufacturing facility. So what we did include was fuel loading. What we did not include was testing in the factory. And I'll get to that in a couple slides.

DR. BLEY: Okay, thanks.

MR. RECKLEY: So if we go to slide 33, we did include in the proposed rule as released for

comment, proposed 53.620(d), for fuel loading which both authorizes the loading of fuel into a manufactured reactor and sets some requirements for that. The first requirement is that we're only at this time addressing fresh or unirradiated fuel. And that would be loaded into a manufactured reactor.

And we would use the provisions of Part 70 on the control of special nuclear material to basically lay out what the requirements were for this activity. One of the key provisions was that 620(d) includes a requirement for two independent physical mechanisms to prevent criticality. And another important part of this proposed section is that the Commission -- it includes a Commission finding that the manufactured reactor module in the configuration with those critical prevention features in place is not in operation.

And this is just -- reflects that historically if you look at the difference between the construction permit, the operating license, the difference between, in Part 52 when we make the 52.103(g) finding, those are keyed off of loading of fuel. And so we wanted to include a provision that says something different. You can load fuel. And as long as these criticality prevention features are in

place, you're not in operation. Dennis?

DR. BLEY: Yeah, I was just thinking about spent fuel and the casks. And NRC's role in certifying the casks is that any protections built in the casks have to survive the drop test which were pretty severe. Reactors moving across the country could run into the same kind of problems. How does that apply here?

MR. RECKLEY: Well, we're addressing in this section the conditions in the factory or in the manufacturing facility. For shipment, it would still need to meet the requirements of Part 71 which includes what you just mentioned. So --

DR. BLEY: Okay. So it'd be essentially the same as for a spent fuel cask.

MR. RECKLEY: Yes, or fresh fuel as it's being shipped.

DR. BLEY: Or fresh fuel, sure. Okay. Thank you.

MR. RECKLEY: Okay.

MEMBER PALMTAG: This is Scott Palmtag. I've got a question about this. So normally in a reactor, you load the fuel and you have fuel power physics testing to kind of confirm that it's loaded correctly because you could have the wrong enrichments

or wrong BPs. How are you going to confirm that your fuel is loading correctly if you don't allow testing? I mean, it could be loaded incorrectly and then shipped across the country in a wrong configuration.

MR. RECKLEY: Well, and that's part of the argument to allow testing in the factory. And I'll get to that in a couple slides. Short of that as we've addressed it in the proposed rule as it was released, you would do as much of the verifications as you could.

And that could include both visual and procedural controls, any non-nuclear testing that you could do, or using non-criticality type testing, so even if you were using radioisotopes to support some of your verifications. But again, we stop short. The additional testing would need to be done then at the site, the final place of operation.

That's where -- under what we have currently, that's where additional zero power physics testing, confirmations would need to be done. As you're alluding to, that could introduce problems because now you're shipped it. You've installed it.

Now you find an error. And for small microreactors, it might be problematic then because the site to which you're shipping it may be a mining

facility, some remote location. May not want to have full capability to do a reconfiguration or reloading.

And so that's the reason some people have asked for factory testing. But again, we recognize that. And if we go actually to the next slide, this just has some additional things.

It's basically the same as I've already mentioned. So we can go to the next one under slide 35 -- yeah, there we go -- that we included per the directions in the SRM a question on factory testing. And so this is the question that's included in the -- this is the direction in the SRM.

And then in response to this, we included a question in the Federal Register Notice. And then we also provided -- and this goes to Dennis' point -- a white paper that gave some initial thoughts on how this would work. And as you can imagine, it adds additional complications when you're proposing to load fuel, do testing, then restore a manufactured reactor, then ship it, and then install it at the final place of operation.

From a licensing perspective, you're not only dealing within special nuclear material and the manufacturing license. You're dealing now with an operating license and the introduction of byproduct

material. And so what we did in the white paper was to lay out how we thought this might work.

And I wasn't going to get into detail in the white paper. We can if time allows, maybe afterwards. Because the white paper, we only issued to solicit comments to see if this was a feasible approach to help people think through all of those interrelationships between parts.

And the comments that we received on that might be it will work and then we'll take that up with ACRS as we go through the comment resolution. But we also might get a comment or comments from potential users that say it was a nonstarter. So it wouldn't be of much point to go through it today.

So if we go to the next, slide 36, this is the question that we included in the Federal Register. And basically it is, should we include provisions? And as we've heard, there are some practical arguments as to why it might be a good idea to allow that kind of testing.

One of the things we were trying to get to in the question in the Federal Register and also in the white paper where we gave a possible approach is, what would be the limits on operation? Could it be done where you try to do it at low power levels in

short durations? Because the more operation occurs, the more byproduct material is introduced.

That introduces obviously shielding questions. It raises the complexity of transport and installation at the final place of operation. So, how much could we limit operations associated with this testing in order to minimize the amount of byproduct material that would be introduced?

If you set out these kind of controls, limited power levels, loading with fresh fuel, then what current limitations on operations in Part 53, the proposed Part 53, might be relaxed? For example, when we had talked a little earlier about aircraft impact assessment, if it's fresh fuel, if it's limited such that there's limited byproduct material that has not only release category -- I mean, the inventory, it also is going to determine things like how much decay heat. How much heat removal might I need?

But if I appropriately minimize all of those operations, might aircraft impact assessments not be needed? Might I revise even other external hazards? So this is a question we ask in the FRN. Go to slide 37.

One of the important areas that we asked is within the requirements, what would be the role of

the actual manufacturing facility? And if the manufacturing facility is playing a key role, then how would it interplay with things like the definition of construction, the operating requirements under subpart F, the personnel requirements under subpart F? So again, just to have people think through, on these reactors that might be loaded and tested, what is the role of the factory itself?

Would the testing be done in a special area that would have the ability to isolate that area? Maybe the ventilation system then becomes a design feature that's credited in the licensing basis events. In addition to the manufactured reactor, you would have the ventilation system associated with the testing room. So again, we're just asking people to think through how all the puzzle pieces would fit together.

MEMBER HALNON: Bill, this is Greg Halnon. Did that puzzle go all the way to potential decommissioning of the manufacturing facility?

MR. RECKLEY: Yes.

MEMBER HALNON: Okay, thanks.

MR. RECKLEY: And in the white paper, we kind of addressed this with some various thoughts on how that might be done and whether subpart G for a

reactor facility is appropriate. Or maybe it could be done more in line with Part 70 of a fuel cycle facility. So all of those questions, again, this just raises -- it basically brings all of the NRC reactor regulations into play and everything needs to be thought about, all the way through decommissioning, so yes.

Then lastly on this slide, what licensing mechanisms should be used? There's different approaches. The staff put out a paper a year or two ago on microreactors that brought up some of these various possibilities. Every manufactured reactor could receive its own license. So we could do this individually.

There's also the capability to issue a single combined license that would address the manufacturing facility and it could support multiple manufactured reactors. So kind of -- this is the approach that we modeled in the white paper that we released. So one COL would be issued, and then there would be a process but not relicensing for each manufactured reactor.

And then how to handle inspections, tests, analyses, and acceptance criteria or ITAAC, keeping in mind that Part 53 adopted much from Part 52. And so

manufacturing licenses would include ITAAC. And it makes sense that those ITAAC would -- some of those ITAAC would need to be addressed if you were going to do operational testing in the factory.

MEMBER HALNON: I think it's interesting. I'm glad to see you're asking the right questions. Just in my mind, I find it hard to believe that you would not want to test this before you send it out.

I mean, like you said, if you manufacture this, you send it out into the field into a remote mine, you try to start up, it doesn't work. What's your options? Ship it back to the factory? But I'll wait and see what you come up with. Thank you.

MR. RECKLEY: Thank you.

DR. SCHULTZ: This is Steve Schultz. I agree that this may have utility for certain reactor types. You've really laid out a pretty good draft of what will be the requirements moving forward for it, what you presented here and what is in the detailed descriptions which you prepared.

It just will be interesting to see what utility of the testing within the manufacturing facility will have for different types of reactor designs. I can see if for a microreactor. When you're looking for SMRs with nth of a kind type of

design expectations and commercial expectations, it may not be useful.

MEMBER HALNON: This is Greg. It seems like the amount of time that you would be in, quote, operational mode, versus the amount of time that it's going to be on site operating, your risk numbers are going to be different. I mean, you have to think, to me, that period of time, if you're going to operate it for hours and it's got a 40-year life, it seems like the risk of it might be low. I don't know. So it seems to me this needs to be looked at from a risk informed perspective as well, not just all the regulations apply. And some may not apply just because of the low frequency potential.

MR. RECKLEY: Right. And we did try to address that by saying the assessment, the safety analysis, the risk assessment done for this operation within the factory should reflect the manufactured reactor as it's being tested. So that would mean low decay heat, for example. If you have low decay heat, then you have associated much reduced requirements for heat removal system, right?

So it is possible that they could go through this. And basically the identification of possible malfunctions and events would result in there

not being nearly as much in terms of what needed to be controlled and give them additional flexibility in comparison to the long term operation at the place of operation. So yes, we'll see.

And to Steve's point, this probably is a niche, right, for SMRs that are going to be routinely refueled, whether that would be anything that would be done in the factory versus done at the site since you're planning to do refuelings anyway. That may not really come into play for many reactor designs. But for the microreactors, this is a potential niche. So if there's no more questions on this, I'll go to --

MEMBER PALMTAG: This is Scott Palmtag again. I agree. That really is for microreactors. If you have an SMR where you're going to be able to refuel it on site, you can do more testing there.

But this is would be for microreactors. What Greg said is correct. I mean, you're going to have no dosimeters. There's basically going to be no dose or decay heat.

But one way around it would be require the manufacturing facility to have a site license. I think that would cover it. It would be interesting to see what you come up with.

MR. RECKLEY: Yeah, and so far, the white

paper reflects that, the low dose considerations.

MEMBER KIRCHNER: Bill, this is Walt Kirchner. Forty years ago, I had to think through all these issues. I think what we determined at the time, we were looking at a first of a kind and then 13 more microreactors.

And I'll just remind people that once -- depending on how these are transported, you pretty much will have to repeat all your startup testing at the actual deployed site. So our assessment at the time was to approach criticality but not build up any significant inventory because it turned out that shielding considerations and transport and the size container -- certified container needed pretty much dominated the logistics considerations. And then as you earlier pointed out, if it indeed is a very modular small system like a microreactor, then you've got to be concerned about exposure personnel at the site when you erect it once it comes out completely intact from the manufacturing facility.

But you would still have to look at the transportation loads that the module was subjected to. And so you would pretty much, I think, find yourself repeating your ITAACs as you point out. So just an observation.

I mean, obviously, if economics weren't an issue for microreactors that were going to have a large number nth of a kind, then having a prototype and doing all of this and then actually replicating that prototype, you wouldn't need to do further testing. You'd just do your kind of startup testing once you're at the deployed site for the nth of a kind. You wouldn't do it with each individual one other than the quality checks that you would do in the assembly phase.

So just an observation. I think Scott's observation is also appropriate here. Probably most SMRs of any size are going to have to design for refueling and decommissioning. And that's also a consideration for the microreactors, depending on size and shipping container availability that you may have to disassemble in the field to deal with the spent fuel. But it's likely that most SMRs of significant size would have to design for refueling.

MR. RECKLEY: Yes, thank you. And a good point that you made in there is that the ITAAC would -- there would be ITAAC at the final place of operation. And that would support the actual operating at that location.

And to whatever degree it needed to also

verify that nothing was adversely done during shipping. So if we go to slide 38, I think I want to turn this over to Jesse and give him a chance to talk about the unique operating staffing and operation stuff that we put into Part 53. So moving from subpart E to subpart F on operations, subpart F is divided basically into three parts.

And that is the configuration control for the plant equipment, making sure they have the capabilities and availability and reliability that was established through subparts B and C. Then there's the personnel, and then there's the plant programs. And so there in the bottom, it basically says the first part is addressing the plant equipment.

The middle part that Jesse is going to talk to momentarily is talking about the plant personnel. And then the last sections in subpart F, 845 through 910 are dealing with the plant programs like radiation protection, quality assurance, emergency planning, security, and those things that we're not going to talk about today just because they didn't change very much. But we did want to refresh everyone on the personnel requirements. So if we go to slide 39. And Jesse, if you're around, if you could take over.

MR. SEYMOUR: Thanks, Bill. I appreciate it. My name is Jesse Seymour, and I'm an Operator Licensing Examiner, Human Factors Technical Reviewer in NRR.

And I'm going to be providing a relatively brief overview of subpart F's provisions for personnel and human system considerations. These span 53.725 through 53.830. And I'm just going to be highlighting some of the more significant elements of those sections.

At a high level, I'd like to say that most of the substance of these sections did not change in response to the SRM. We had previously integrated Framework A and B under this particular section. So the same requirements applied with respect to the framework.

But we did have to go through and make some editorial changes to pointers and so forth and also remove provisions for AERI. However, there are some elements in here that garnered a lot of interest by both the committee and stakeholders. So I do want to highlight those as we go through.

Again, this will be relatively brief. So Sections 53.725 begin with some general requirements that apply to facilities. These include content of

applications requirements. The key things I'd like to point out, human factors engineering is approached differently under Part 53.

And there's a focus on where humans are involved with the fulfilment and support of safety functions versus generic applications or a control room. Additionally, there's a facility-specific staffing plan requirement that's employed which instead of having a prescriptive staffing level akin to what we see in 50.54 currently. Instead, it looks at an approved staffing plan that's been supported by human factors engineering insights and also points to engineering expertise.

And these are items that I'll go into a bit more detail on the next two slides. Conditions of facility licenses are covered under 53.740. Again, this is somewhat analogous to 50.54.

And some key provisions we introduce there are provisions for automatic load following by plants with some restrictions, additionally provisions for the oversight of online refueling as well. Part 53 contains standalone frameworks for the licensing of senior reactor operators and reactor operators. And within that framework, we introduce a number of new flexibilities.

So this includes the use customized Commission approved operator licensing programs, both through the training and examination of operators by facilities. Again, this is a way to have right sized and technology inclusive programs that focus on what those operators need to do based on the designs of the plants, both the human role and safety. Additionally, it allows for facilities to administer licensing exams with the presence of the NRC and also with NRC approval of the exams themselves.

Additionally, Part 53 introduces provisions for a new type of licensed operator. This would be the generally licensed reactor operator which would be a departure from our history of only specifically licensing SROs and ROs. This type of operator would only apply at a limited set of facilities that we refer to as self-reliant mitigation facilities.

And these are facilities that by virtue of meeting a set of criteria have determined to be of a design such where humans do not have a significant role in the fulfilment of safety functions or in achieving the safety outcomes that the plant needs to. So again, the general license for those operators is contained within that section. And I'll touch upon

that in the latter slides as well.

Lastly, plant staff training requirements covered under 53.830 which is essentially a modernized version of the training rule that we see under 51.20. If we could move to the next slide, please. So digging into some of the highlights that I touched upon from the previous slide.

So under 53.740, load following would be permitted provided the plants have appropriate design considerations that are built into it. So these are touched upon at a very high level in the rule. We do tend to expand upon this further within guidance documents that we're working on developing under the content of application regulatory guidance under development.

At the level of regulation, what we would essentially mandate is that one of three measures has to be in place to keep the plant from departing from acceptable operating regime during load following. And that would either have to be an automatic protection system that's dedicated. Again, something separate from the credited reactor protection system, and able to essentially truncate those transients before you get to an RPS actuation set point.

The use of an automated control system, so

again, the use of automation to manage plant systems and to limit transients caused by load following such that you don't depart from the acceptable operating band. Lastly, something more akin to what we see utilized over in Europe, the continuous oversight and ability for immediate intervention by a licensed operator. So again, that operator that's there to arrest that transient and to take manual control should it be necessary.

Another modification that we make is under the 53.830 plant staff training set of requirements, what we do is we depart from the past practice of having prescriptive time frames where those programs have to be in place. So right now, there's an 18 month timeline where you have to have a SAT-based training program in effect for plant staff. What we do instead is we update that to instead be marked off of when those personnel are needed to support plant operations.

So again, when you get up to the point where you're getting ready to bring that plant online, when you're getting ready to begin operational testing and so forth, that's the milestone, right, the need date that drives when those personnel programs have to be in place. Additionally, what we do is we go

through and we are more flexible in the personnel categories to reflect that there can be some unique roles and assignment of tasks to personnel in nontraditional manners at advanced reactors. So again, combining roles, operations, maintenance, radiation protection, at facilities that have small staff and complements, we wanted a rule that was capable for accommodating that.

Under 53.730, staffing plans will be proposed by applicants. Again, so in lieu having us establish a prescriptive staffing requirement at the onset, staffing plans would be submitted to the NRC supported by agency analyses and performance based tests. So again, this is a very similar process to what's currently done under NUREG-1791 to justify exemptions from the prescriptive staffing requirements.

But this would be the starting point under Part 53. Once we review and approve those staffing plans, that would become the condition of a facility license. New staffing plans would establish the operator numbers qualifications and locations of the personnel that are needed to fulfill plant safety functions.

Additional features that are included in

the staffing requirements are a flexible requirement for engineering expertise that is used in lieu of traditional shift technical advisor staffing. Importantly, and this is articulated under guidance that we have for staffing that was released as part of the proposed rule package, so again, the DRO-ISG-2023-02 documents for staffing plan reviews. The engineering expertise would not necessarily need to be co-located with the plant.

It could be remotely located if there's provisions for the receipt of data and for communications with the facility and for having those individuals in place within a ten-minute time frame which is akin to current SDA practices. Additionally, those individuals would be capable of covering more than one facility provided that they have the requisite training and familiarity and plant data and procedural access needed to do that. Also, we introduced the location neutral approach to operator staffing.

So the results of the staffing plan will allow for facilities to provide justification as to where those operators should be located. Yes, Mr. Kirchner. I think you have a question.

MEMBER KIRCHNER: Thank you, Jesse. I

didn't mean to interrupt. But could I go back to the -- just clarification on the first main bullet. The first sub-bullet 1 says an automatic protection system.

So when we think of a reactor protection system, we're usually thinking of the safety-related standalone system. So the first one makes sense to me. The second one suggests that since you used or, this automated control system, what kind of quality would be required of that?

Would it be equivalent to a reactor protection system? Would it be vulnerable to the internet and cybersecurity concerns? And then the intervention of the RO, SRO, GLRO presumes that there's always a person at the control system. So just clarification is what I'm looking for. The middle bullet is the one that's of concern.

MEMBER ROBERTS: This is Tom. Before you answer, I have a similar question. I was going to expand that. It seems like it's got an issue of separation, control, and protection because you would have known that a control system if you had load following that was commanded by something other than inherent characteristics of the reactor.

And so that control system could be the

cause of the command to go to excessive power. And so it would seem like you would need one if you had two or at least the capability of one built into two to credit the automatic control system as capable of stopping what might be the source. Thanks.

MR. SEYMOUR: This is Jesse. And I'll provide the discussion of this. And again, there is some complexity here. A key item that I want to point out is that what we want to do is make sure that load following which is, I think, a good way to put that is it's an operational nice to have, right?

It's not allowed to interfere with protection which is something that must be there. So again, we don't want load following to credibly be challenging the reactor protection system. We want other features to be in place that are going to prevent a reactor protection system from being the bumpers on this process.

So when we see an automatic protection system under one, what we were envisioning there is something kind of akin to what we see on generators at facilities right now. And what you'll see are certain features that are there to protect overexcitation of the generator. In some cases if, you know, reactive load conditions on the grid get to a certain point, an

automatically set generator voltage regulator can go into a regime where there's, you know, overexcitation concerns, potentially overheating the machine.

And what you'll have is protection systems that will kick in. Again, these are not on safety. This is on the balance of plant that will kick in. Essentially, if that is not corrected within a certain period of time, try to limit how far you can go with an excitation.

And if it's subsequently not corrected, to go ahead and trip the generator itself. Now obviously that induces initiating events. And the reactor protection system may respond to that as needed if you're above a certain power set point, the limitations it steam dumps.

So in this discussion under one, what we are seeing from my perspective was that this would be more of a non-safety, you know, kind of deterministic set point built into certain balance of plant components that would essentially truncate that transient, right, not let you go further. But that was the paradigm we had in mind with one. With two, with automated control systems, again, when you look at the ability of the control system to potentially comprehensively manage output of the generator to

manipulate reactivity control systems which is something we've seen in existing plants where certain plant designs have automatic rod withdrawal and so forth to maintain temperature.

In that case, what we anticipated there is that the control system would receive for all intents and purposes a request from the load dispatcher or if this is the single source of power on a microbit, for example, you know, input from existing load conditions, that would come in, in the form of a -- again, for all intents and purposes, a request to the system that the system would then have to determine if it could meet. And if it could, you know, what the ramp rate would be and so forth to go through and to meet that safely.

So again, this was not meant to enable the non-licensed load dispatcher, you know, on a grid to send demand to the plant and just begin moving the generator under two. What it was meant to do is essentially put a middle man there for lack of a better way to put it that would receive that request and then it would go ahead. And if it was able to meet that, to go ahead and implement it and then, in many regards, acting as a surrogate for the operator.

But this is more of a control function

versus a truncation. Under one, what we are envisioning is that the load change request would come into the plant. And the plant would begin to respond to meet it.

Again, it would be sort of a direct control of the power generation side of the plant with protective features that would truncate that before you reached RPS. Two would be more of a fine control where a request comes in and the plant implements that acting as a surrogate for the operator with three being the operator acting as that middle. In all cases, there would always be the normal reactor protection system in effect, you know, as required by other plant design characteristics.

So again, and we can revisit that if there's further questions on it. But I'll go ahead and I'll move on with some of the other bullets here. Prescriptive time frames being used to establish training programs, as I mentioned earlier, that's another modification that we've made.

And also, you know, as we go through and we talk about the staffing analysis as mentioned, there are other improvements that we've looked to make and other flexibilities we've looked to introduce. So again, the flexible requirement as I mentioned earlier

for the shift technical advisor, the location neutral approach taken to operator staffing. Again, if fulfillment of safety functions is something that humans have a role in doing at a plant, there's going to be a driver where the onus is going to be on the applicant to demonstrate where that needs to be achieved from.

And then it can credibly be achieved from that location. So again, if there's a proposal to not have a control room and to, say, control from some alternate location in the plant, that's all going to be factored into the human factors engineering analysis and the staffing plan validation that's submitted to the NRC. Lastly, when it comes to self-reliant mitigation facilities, these are facilities where we don't envision that the human would have a credible role in the fulfillment of safety functions.

That by virtue of the plant design, the plant itself would be largely insulated from the influences of operator performance. And so there, you know, on the basis of these plants as the name implies are self-reliant from a safety standpoint without the human role. We would not be necessarily interested in the staffing being adequate to fulfill safety functions since the humans would not have a role

there.

So here what we do is we step away from that HFE-based staffing analysis requirement. And instead we have simplified requirements that really just pertain to the administrative oversight of a GLRO and also to have that engineering expertise requirement available as well too. If we could move on to the next slide, please.

Okay. And again, so this is my final slide. And here I wanted to dig further into the generalized reactor operator area as this has been an area of interest. So I mentioned the self-reliant mitigation facility which would among other things be eligible to be staffed by these generally licensed reactor operators.

Beyond that, these facilities would also have significant modifications to the application of human factors engineering and, you know, what's required for the operator licensing program, again, all respecting that there's a diminished role in the fulfillment of safety by humans there. So the criteria -- and again this is a simplification -- to achieve this would be that, you know, no human actions could be necessary to meet radiological consequence criteria as discussed by Anders earlier to address

licensing basis events or to provide for adequate defense in depth. Additionally, safety functions could not be allocated to human action.

And lastly, there would have to be reliance upon robust, highly reliable safety features. So here we'd be looking for safety features that aren't going to be subject to human failures, again, errors of omission and commission. And the most kind of readily available examples of those would be things that are inherent or robust passive nature, you know, in nature.

However, what I would also say is that this doesn't exclude the possibility for different types of passive features or potentially even active features to be used, provided that there's a demonstration of how those are going to be robust enough to not be subject to those human failures. So again, engineer measures to enforcing performance and so forth. Yes, Member Bley?

DR. BLEY: Yeah, Jesse. Just as you go through this list, things seem almost a little inside out to me. Visually, I think of designers coming up with ways they want to do this and submitting with you folks reviewing it.

It seems like you're getting a big jump on

thinking of different ways this would work, I don't know how thorough you've worked through all these ideas. Have you gotten additional ideas from designers and maybe potential operators of ways this could work? Or is this pretty much all things that have thought through at least to some extent by the staff?

MR. SEYMOUR: At this stage, we have received some input along the way that was informal during the course of stakeholder actions. But at this stage now that we have the proposed rule out for comment, we're expecting kind of the main body of feedback to come in. So what you're seeing here and what we've discussed previously was largely generated by the staff with regards to the self-reliant mitigation facility.

And a lot of that was a function of this being a new class facility, new type of operator and so forth. What I'd like to say is that this is really the synopsis of the rule language here. And the rule language itself tries to be performance-based and tries to leave the door open to how this could be met by a variety of means.

As I mentioned, the safety features used to meet this aren't necessarily limited to being

inherent or passive. They could potentially be active. But again, we'll leave it to the designer to make that case for how these are credibly not going to be subject to those human failings.

And the reason being is that under this type of framework, we're not even staffing these facilities potentially with individuals that we would tend to credit to take those actions. The GLRO, for example, would not be an individual that receives a medical exam. So, could there be a medical issue that incapacitates them when they take an action potentially.

So we don't want them to be credited. So I would say that we are expecting significant feedback from the industry on different thoughts on this. There has been some allusion to that from different stakeholders that we've talked to. And we're definitely receptive to different ways we could go about this.

But fundamentally, the self-reliant mitigation facility is just one kind of subtrack within Part 53. If a facility does not meet the threshold to be considered a self-reliant mitigation facility, they still -- even staff by SROs and ROs have access to flexible staffing requirements,

tailored operator licensing programs that they can modify to meet their specific needs and so forth. Just great flexibility as improvements over Part 55.

So I just want to point out that, you know, it is a high bar to be considered within this category. There are a lot of kind of loosening of the regulatory footprint and so forth happens when you get there. But just because a facility doesn't screen doesn't mean that they don't potentially see a lot of benefits, I think, and improvements over the existing framework.

DR. BLEY: Okay. Thanks. That's pretty good. I guess the thing that I'm not fully comfortable with yet is given we've written this into the regulation, it kind of says these things are actually reasonable and could be licensed. And I just wondered if you really thought that fully through. I know there is current and past members of this committee who've expressed a little skepticism along the lines of self-reliant systems.

MR. SEYMOUR: That's an excellent question. Again, this is Jesse. And what I would say is that the way that we've tried to approach that because this is a new area for us to go into is we try to think through the potential for an autonomous

reactor, for example, where you would not have a human there.

And so you begin to think of what would the machine -- what types of criteria and performance characteristics would the machine need to be able to meet to operate safely in the absence of many opportunity for human intervention, right? Because no one is there. And as we thought through that, what we began to see is that different measures all have their own vulnerabilities.

Again, if you look at a measure that was strictly based on what was credited within PRA, well, what if the PRA methodology was deficient? If you looked at mandating that it had to be reliant upon only inherent safety characteristics, what if there was some analytic uncertainty in some of those inherent characteristics such as the construction of the fuel in the first of the kind build and so forth? So what we did is we used a set of five criteria that has some -- it has some synergy between them but at the same time have enough independence such that you are providing that ability to have some resilience against some of those vulnerabilities.

And it looks like we jump back in slides here. So we'll catch back up. But again, it is an

area where we've tried to tread carefully. What we've tried to do is to envision where we see the technology going over the long term which is, again, you know, a greater drive to inherent safety, a greater drive towards remotely located facilities, autonomous reactors. And then to say what are those performance characteristics that -- and again, I use the term performance there very, very explicitly because we're looking at it in a performance-based way, right? But what are those outcomes you need to be able to achieve to do that safely?

DR. BLEY: Yeah, thanks, Jesse. It'll be interesting to see where we end up here.

MR. SEYMOUR: Okay, yeah. Thank you. So again, just moving on and going through some of these other areas, 53.805 talks about facility license requirements for GLROs. So again, here very different than ROs and SROs. The license would be located within the regulation.

It is truly a general license. That doesn't mean that there isn't the capability for individual enforcement. There absolutely is, and that's written as a condition.

So there would be the same avenue the NRC could take to take action against an individual who

does something inappropriate or even to bar them from being able to operate under that general license. That's all built in there. Additionally, the facility would be on the hook for implementing the training and examination programs.

And we envision that those activities would still be subject to NRC inspection. And in all cases, they would be subject to NRC approval. Under -- again, I talked about the general license for GLROs under 53.810 as well.

Under 53.813, in a similar manner to the SRO and RO training programs. GLRO training and exams proficiency would be customized based on facility needs submitted to the NRC for approval. The NRC would provide approval of those training programs and of the examination programs as well too.

And then once that was done, you know, the facility would be responsible for administering both of those. Now the facility, because of the requirements of 53.805 would have some regulatory hooks that would hold them to the appropriate implementation of those approved programs. So again, that would enforceable.

But here with the GLRO plant, we envision that the NRC would largely assume an inspection role

of coming in and confirming that those programs are being implemented versus coming in and looking at the individuals for their examinations on an individual basis like we currently do. And again, this is just for the sumps of self-reliant mitigation facilities. And lastly, 53.820 just deals with the cessation of this.

As a general license, it would only pertain to individuals that are employed at that facility. So that concludes my slides. We can transition on. But I just wanted to pause and see if there are any further questions from the committee.

MEMBER HALNON: Jesse, this is Greg. First of all, I think where you are now does reflect this evolution from the certified operator to where we are now. I'd appreciate you taking a lot of our comments and kind of modifying them as necessary but working those thing in such as the STA and some of the training and some of the training issues that we brought up.

One question I have, I wasn't sure about is we had recommended that there be -- it was back in certified operator. But we recommended that there be at least one person with the -- that everybody reported to that had the same level license, the GLRO.

It is in your thinking that in the fleet of GLROs there would still be one person that would be accountable for the license aspect of this? Right now, usually the office manager has an SRO or inactive at best but they're licensed.

MR. SEYMOUR: Yes, and that's a feature that we see appear in the administrative section of tech specs which is driven if memory serves me by commitments to -- I believe to Reg Guides and ANS standards to maintain that senior license position. So again, if we look at standard tech specs for plants, that's where we'll see that. And we see it emerge there.

What I'll say, that's an item that we did consider as we went through and looked at this. And what we -- where we fell out on that was that it was something that it currently doesn't exist at the level of regulation under the existing framework. So consistent with that, we didn't want to introduce that at the level of regulation here within this framework.

Now again, it still does appear. It appears as a commitment for other plants. But our thinking had been that was something that would be better served by existing at the level of guidance to keep it consistent with where things currently are.

Again, not taking something that's currently at the level of guidance and then emerges as a commitment and elevating it to the level of regulation. But that's definitely an item. And I know exactly what you're talking about.

There's an operations manager and an assistant operations manager. And generally what we see in tech specs, in fact, universally, I think, is that there's a commitment that one of those two will hold a senior reactor operator license for the plant. And that puts them into this role we refer to as senior license holder.

And it grants them kind of a unique position where they can be the final say on some of those technical debates that you get into in the SRO role and serve authoritatively. So I think it's what you're referring to. And that's something that's still -- it's still on our table to work through the guidance. But that's where we fell out on our side was that it just seemed better served to have that be a guidance matter.

MEMBER HALNON: Fair enough. Certainly could be outlined in the staffing plan. I would imagine you could ask the questions there if you needed to. But again, I want to thank you for taking

a lot of our comments and making it work through the regulation, how you intertwined those was good. So we can move on. I guess you need to wrap it up. Anders?

MR. GILBERTSON: Yes, we can move on to slide 42. And I'll just be very brief. Not a whole lot more to say here other than just outside of the SRM related provisions that we talked about earlier this morning, other changes were made as appropriate to subparts G, H, I, J, and M and their related portions to the preamble to address crosscutting SRM items like some of the ones I mentioned earlier, references to the subpart K and the framework problem. So beyond that, that really concludes the staff's presentation. And I'll hand it back to you.

CHAIRMAN PETTI: Great. Thank you. Members, any other comments?

MEMBER HALNON: I see Vicki has her hand up.

CHAIRMAN PETTI: Okay. I can't see that. Vicki, please.

MEMBER HALNON: Vicki, you're muted if you hear us. Vicki, we don't hear you if you're talking. Dave, apparently Vicki is having some technical problems. We can't hear her. So --

CHAIRMAN PETTI: Yeah.

MEMBER HALNON: -- trying to get her comments on the record somehow.

CHAIRMAN PETTI: We'll give her a minute. I just want to thank the staff again for giving us this briefing and letting us know where they are. I joke that we could talk about Part 53 one hundred times and we'd still generate lots of questions on the 100th time as we did on the first time.

It's just the nature of the beast, I think. But again, thank you all for the discussion. It was good. Vicki, one last time. If not, let's go for public comments. Anyone online who wants to make a comment, please raise your hand and we'll identify you and let you make your comment.

MEMBER HALNON: I don't see any, Dave.

CHAIRMAN PETTI: Okay. Then one last call, Vicki. Is her hand still raised? I can't see it.

MR. BURKHART: This is Larry Burkhart with the ACRS staff. Vicki did log out. I asked her to try to log back in. So give her a minute, perhaps.

MEMBER HALNON: Yeah, she was having technical difficulties like you were, Dave, earlier.

MEMBER BIER: Hi, yeah. I managed to unmute finally. Thanks for the advice, Larry. I had

just one minor comment which I don't expect to be addressed in a big way.

But it's coming back to the issue that I raised earlier and that Vesna and I have discussed about the issue of safety-related versus safety significant. And I've been kind of pondering and mulling over it. And I'm wondering whether when the term is first introduced it would make sense to state that this is essentially equivalent to design basis related.

And I don't expect anybody to change the terminology because safety-related obviously has a long history. And you want to be consistent with other regulations that use those terms. But it might be helpful to just, you know, translate the first time that this is really design basis related, safety items. And there can, of course, be other safety items for people who are new to the whole dilemma. Anyway, that's my only comment.

CHAIRMAN PETTI: Okay. Well, I guess with that, we can close the session. I want to thank everybody again and we'll see you all at tomorrow's subcommittee meeting. Thank you.

(Whereupon, the above-entitled matter went off the record at 12:11 p.m.)

ACRS Subcommittee on Regulatory Rulemaking, Policies, and Practices:

Information Briefing on Updates to Proposed 10 CFR Part 53, "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors"

January 14, 2025

List of Speakers

- Anders Gilbertson – NRR Technical Lead
- Bill Reckley– NRR Technical Lead
- Jesse Seymour - NRR Plant Operator Licensing & Human Factors

Proposed Rule

- 89 FR 86918 (published October 31, 2024)
- https://www.regulations.gov/document/NRC-2019-0062-0310
- ML24095A161

28 Associated Documents

- 89 FR 86918, Section XIX. Availability of Documents
- https://www.regulations.gov/docket/NRC-2019-0062/document?postedDateFrom=2024-10-31&postedDateTo=2024-10-31

White Paper

- Subpart H, DRAFT Section 53.1480 – Combined license supporting testing of manufactured reactors (ML24344A037)

Past ACRS Interactions on Part 53

- 16 ACRS meetings during development of draft proposed rule in 2020, 2021, and 2022
- 4 Interim Letter Reports
- Final Letter Report dated November 22, 2022 (ML22319A104)
- NRC Staff Response dated February 10, 2023 (ML22341A047)

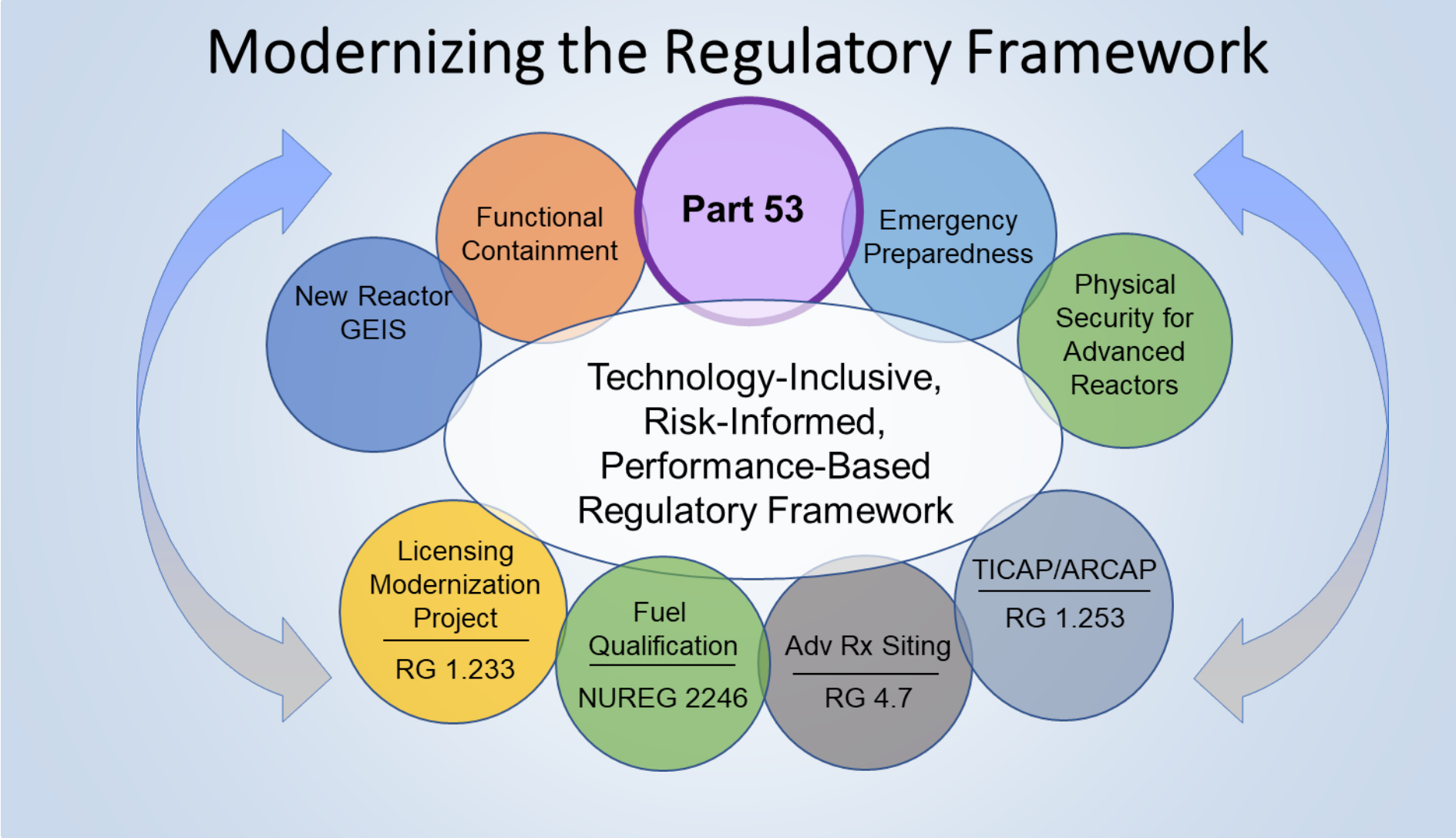
Key Rulemaking Documents

- SECY-23-0021, “Proposed Rule: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors,” dated March 1, 2023 (ADAMS ML21162A093)
- In SRM-SECY-23-0021, dated March 4, 2024 (ADAMS ML24064A047), the Commission approved, in part, the NRC staff’s draft proposed rule with exceptions and clarifications

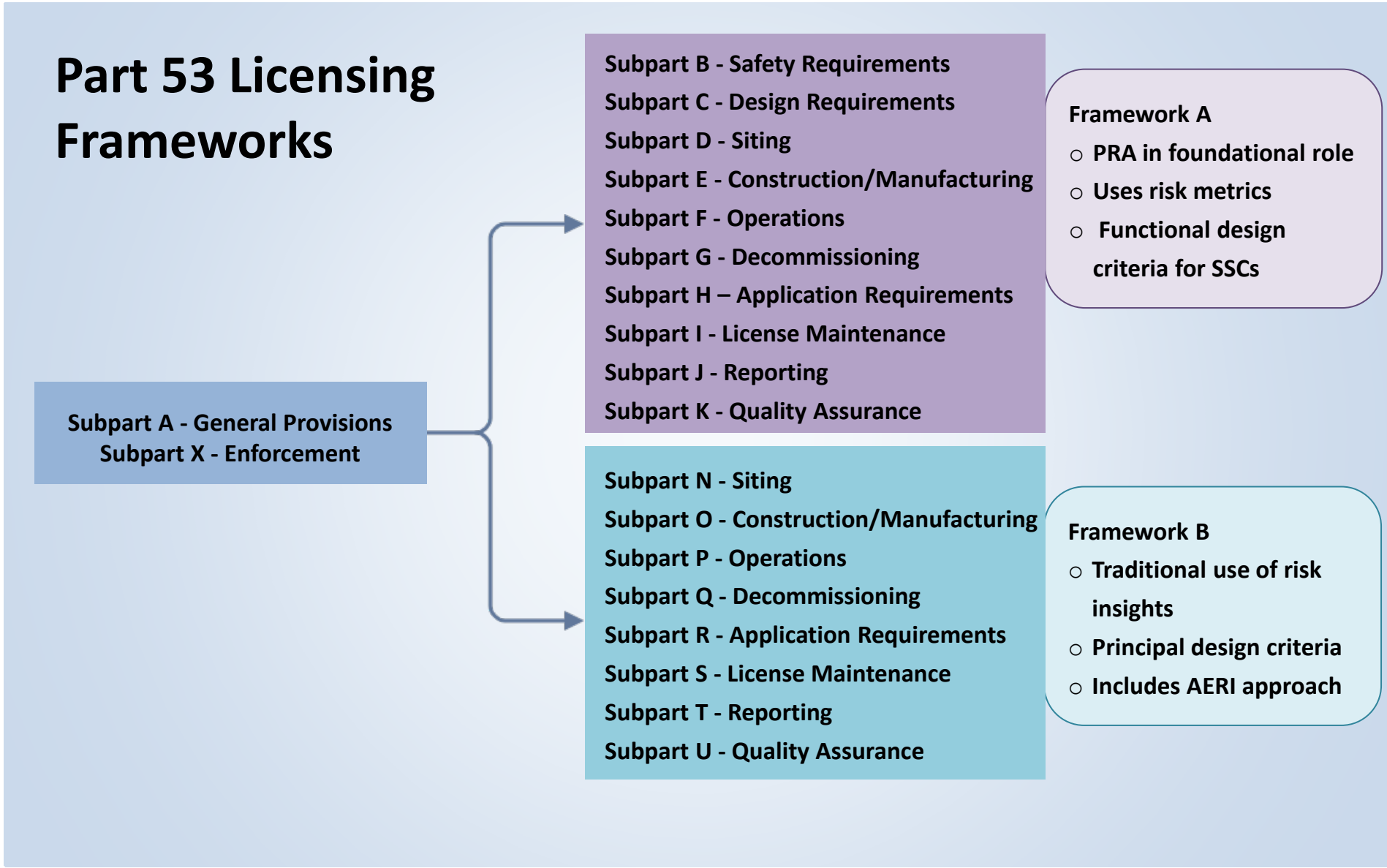
Part 53 Final Rule Milestones

- Close of public comment period: February 28, 2025
- ACRS Interactions: will work with ACRS staff to set up dates for late 2025 and early 2026
- Final Rule to the Commission: May 2026
- Final Rule Published: NEIMA Deadline – December 2027

Recent & Ongoing Activities



Draft Part 53 Proposed Rule (SECY-23-0021)



Staff Requirements Memorandum Summary

1) Remove Framework B from Part 53 and provide new options
2) Replace references to QHOs with comprehensive risk metrics
3) Allow flexibility in PRA acceptability determinations
4) Revise requirements related to as low as reasonably achievable (ALARA)
5) Remove facility safety program
6) Explain process for ongoing evaluations of external hazards in preamble
7) Include requirement for design experience program
8) Include provisions for factory fuel loading and engage stakeholders on possible operational testing of fueled manufactured reactors
9) Harmonize consideration of security-related events within security and emergency preparedness requirements
10) Replace Subpart K QA requirements with reference to Appendix B to Part 50
11) Remove safety objectives section
12) Include question on processes for similar designs at multiple sites
13) Consider suggested edits in Commission vote sheets
14) Provide final version of Federal Register Notice within six months
15) Consider administrative rulemaking for potential errors in Parts 50 and 52

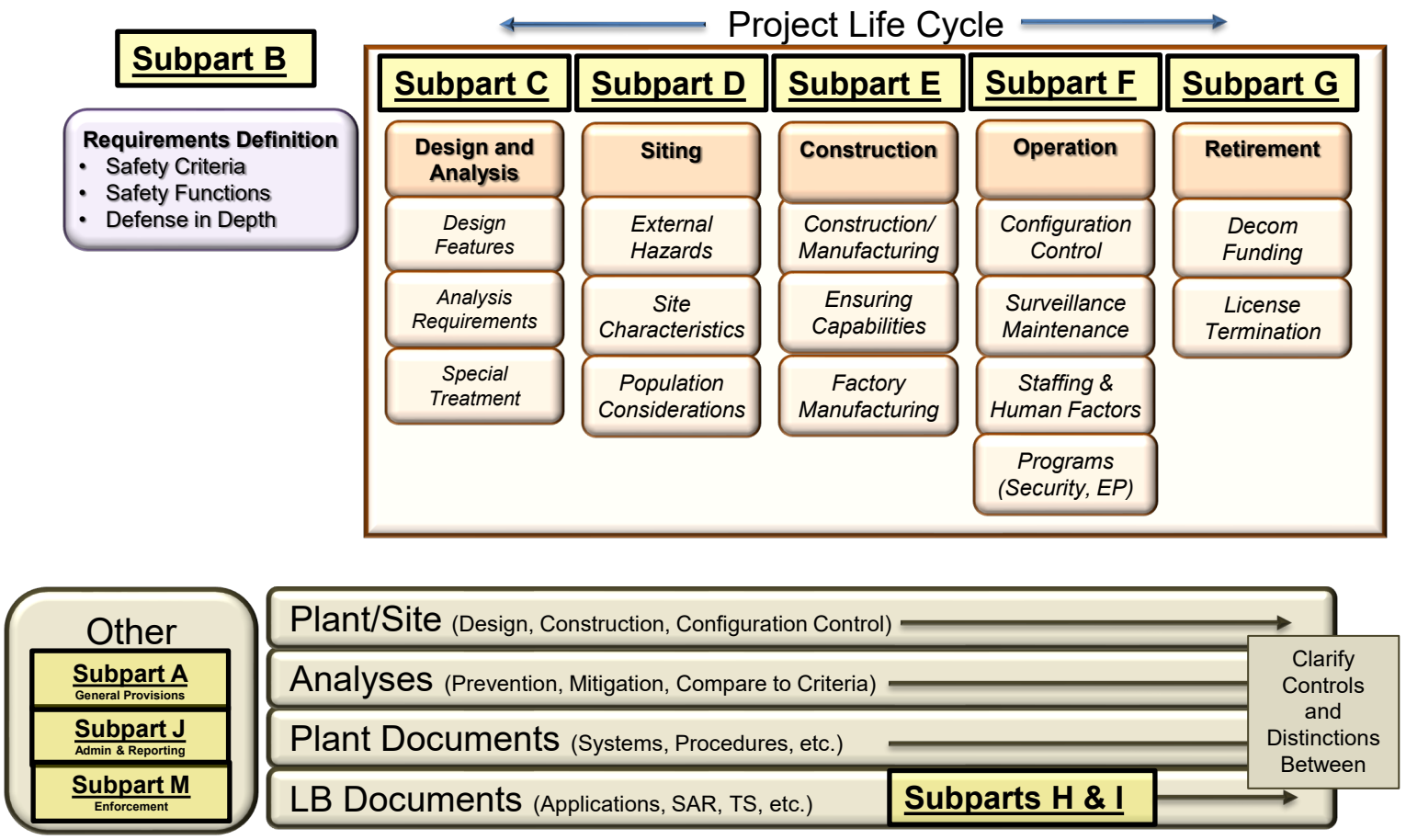
Proposed Part 53 Licensing Framework

Part 53 Organization	
Subpart A	General Provisions
Subpart B	Technology-Inclusive Safety Requirements
Subpart C	Design and Analysis Requirements
Subpart D	Siting Requirements
Subpart E	Construction and Manufacturing Requirements
Subpart F	Requirements for Operation
Subpart G	Decommissioning Requirements
Subpart H	Licenses, Certifications, and Approvals
Subpart I	Maintaining and Revising Licensing-Basis Information
Subpart J	Reporting and Other Administrative Requirements
Subpart M	Enforcement

SRM-RELATED DELTA

- *Framework B and related references removed*
- *Subpart K removed (added references to Appendix B to Part 50)*

Part 53 Structure - Project Life Cycle



Subpart A

General provisions

§ 53.015 Scope.
§ 53.020 Definitions.
§ 53.040 Written communications.
§ 53.050 Deliberate misconduct.
§ 53.060 Employee protection.
§ 53.070 Completeness and accuracy of information.
§ 53.080 Specific exemptions.
§ 53.090 Standards for review.
§ 53.100 Jurisdictional limits.
§ 53.110 Attacks and destructive acts.
§ 53.115 Rights related to special nuclear material.
§ 53.117 License suspension and rights of recapture.
§ 53.120 Information collection requirements: OMB approval.

Subpart A

General provisions

New or Revised Terminology Compared to Parts 50 & 52 (§ 53.020)

- Event categories & related terms
- Commercial nuclear plant/reactor
- Consensus code or standard
- Construction
- Defense in depth
- Functional design criteria
- Licensing basis information
- Safety classification categories
- Probabilistic risk assessment
- Programmatic controls
- Special treatment

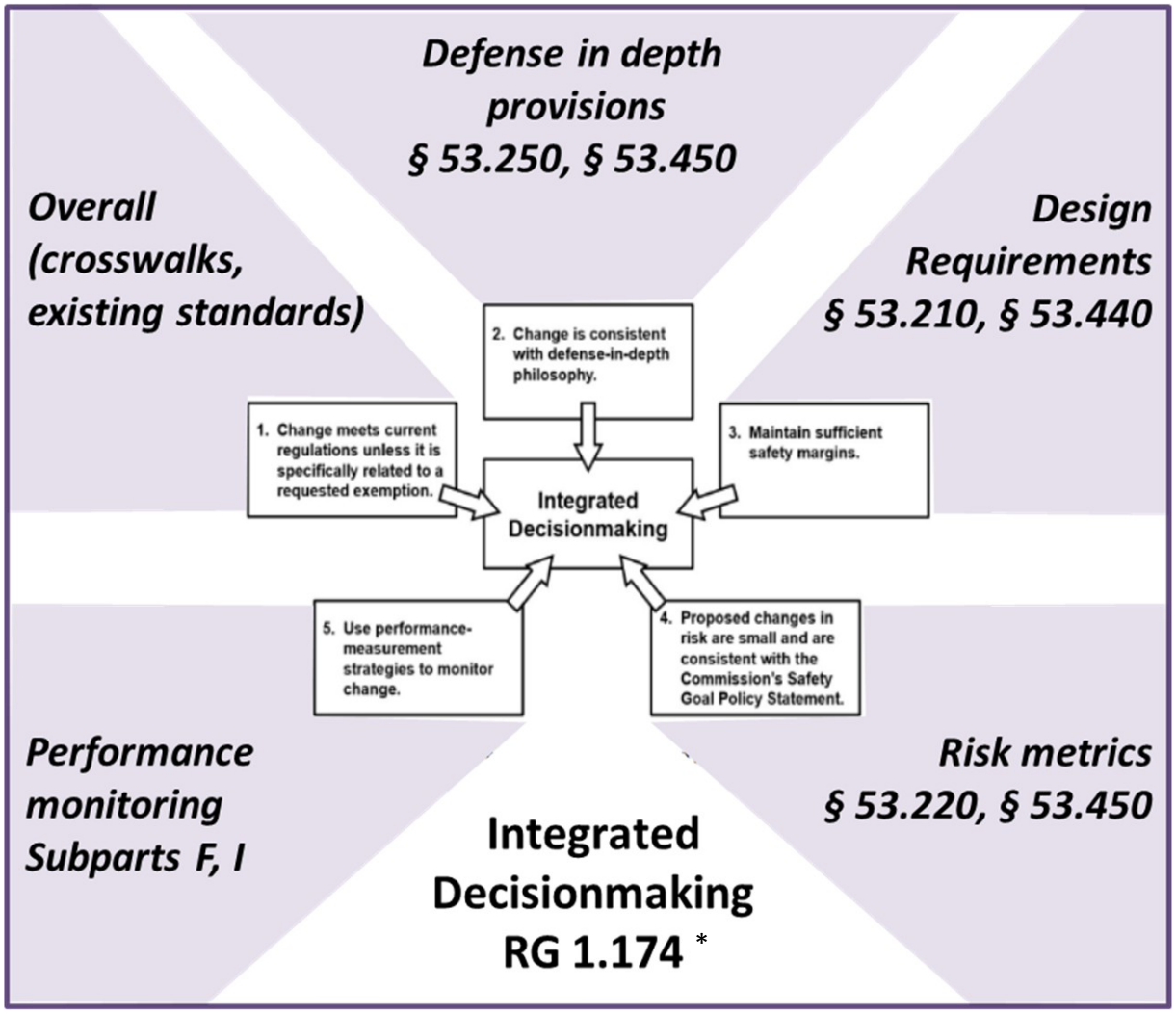
Subpart B

Technology-inclusive safety requirements

§ 53.210 Safety criteria for design-basis accidents.
§ 53.220 Safety criteria for licensing-basis events other than design-basis accidents.
§ 53.230 Safety functions.
§ 53.240 Licensing basis events.
§ 53.250 Defense-in-depth.
§ 53.260 Normal operations.
§ 53.270 Protection of plant workers.

Subpart B

Technology-inclusive safety requirements



*Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

Subpart B

Technology-inclusive safety requirements

§ 53.210 Safety criteria for design-basis accidents.

- Design features and programmatic controls provided such that the identification and analyses of design-basis accidents (DBAs) demonstrate that the calculated offsite doses are below established reference values

§ 53.450(f) Analysis of design-basis accidents.

- DBAs address possible challenges to the safety functions required to be identified by § 53.230 and include events that, if not terminated, have the potential for exceeding the safety criteria in § 53.210.
- DBAs analyzed using deterministic methods that address event sequences from initiation to a safe stable end state and assume only the SR SSCs and human actions addressed by the requirements of Subpart F to perform the safety functions
- The analysis must conservatively demonstrate compliance with the safety criteria in § 53.210.

Subpart B

Technology-inclusive safety requirements

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

- Design features and programmatic controls provided such that the identification and analysis of licensing-basis events (LBEs) other than DBAs demonstrate the following:
  a) Plant SSCs, personnel, and programs provide the necessary capabilities and maintain the necessary reliability to address LBEs other than DBAs and provide measures for defense in depth, and
  b) The analysis of risks to public health and safety resulting from LBEs other than DBAs under § 53.450(e) includes comprehensive risk metrics that satisfy associated risk performance objectives that are acceptable to the NRC and provide an appropriate level of safety.

SRM-RELATED DELTA

Subpart B

Technology-inclusive safety requirements

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

- The analyses must use insights from a PRA in combination with other generally accepted approaches for systematically evaluating engineered systems to identify and analyze equipment failures and human errors.
- The analysis of LBEs other than DBAs must include definition of evaluation criteria for each event or specific categories of LBEs to determine the acceptability of the plant response to the challenges posed by internal and external hazards to provide an appropriate level of safety.
- The analyses of LBEs other than DBAs must address event sequences from initiation to a defined end state and be used in combination with other engineering analyses to demonstrate that the functional design criteria required by § 53.420 provide sufficient barriers to the unplanned release of radionuclides to satisfy the evaluation criteria defined for each LBE other than DBAs, to satisfy the safety criteria specified in accordance with § 53.220 and provide defense in depth as required by § 53.250.
- The methodology used to identify, categorize, and analyze LBEs must include a means to identify event sequences deemed significant for controlling the risks posed to public health and safety.

Subpart B

Technology-inclusive safety requirements

Comprehensive risk metrics and associated risk performance objectives

- Consist of proposed plant risk metric or set of proposed risk metrics that approximate the total, overall risk from the facility and that address the range of possible plant configurations and associated internal and external hazards to the extent practicable.
- The associated risk performance objectives are preestablished, indicative values of the comprehensive risk metrics that are used as part of risk-informed decision-making.
- The methodology for developing and using proposed comprehensive risk metrics and associated risk performance objectives is defined by the proposed requirements for analyses in § 53.450.

SRM-RELATED DELTA

Subpart B

Technology-inclusive safety requirements

§ 53.230 Safety functions.

(a) The primary safety function is limiting the release of radioactive materials from the facility and must be maintained during normal operation and for LBEs over the life of the plant.

(b) Additional safety functions needed to support the retention of radioactive materials during LBEs—such as controlling reactivity, heat generation, heat removal, and chemical interactions—must be identified for each commercial nuclear plant.

(c) The primary and additional safety functions are required to satisfy the safety criteria defined in §§ 53.210 and 53.220, or more restrictive alternative criteria adopted under § 53.470, and must be fulfilled by the design features, human actions, and programmatic controls specified throughout this part.

Subpart B

Technology-inclusive safety requirements

§ 53.240 Licensing-basis events.

(a) Licensing-basis events must be identified for each commercial nuclear plant and analyzed under § 53.450 to demonstrate that the safety requirements in this subpart have been satisfied.

(b) The identified LBEs, ranging from anticipated event sequences to very unlikely event sequences, must collectively address combinations of malfunctions of plant SSCs, human errors, facility hazards, and the effects of external hazards.

(c) The analysis of LBEs must—

(1) Include analysis of one or more DBAs under § 53.450(f);

(2) Confirm the adequacy of design features and programmatic controls needed to satisfy the safety criteria defined in §§ 53.210 and 53.220, or more restrictive alternative criteria adopted under § 53.470; and

(3) Establish related functional requirements for plant SSCs, personnel, and programs.

§ 53.020 Definitions.

- *Licensing-basis events means a collection of event sequences considered in the design and licensing of the commercial nuclear plant. Licensing-basis events are unplanned events and include anticipated event sequences, unlikely event sequences, very unlikely event sequences, and DBAs.*

Subpart B

Technology-inclusive safety requirements

§ 53.250 Defense in depth.

(a) Measures must be taken for each commercial nuclear plant to ensure appropriate defense in depth is provided to compensate for uncertainties in the analysis of the safety criteria such that there is reasonable assurance that the safety criteria in this subpart are met over the life of the plant.

(b) The uncertainties that must be addressed under paragraph (a) of this section include those related to the state of knowledge and modeling capabilities, the ability of barriers to limit the release of radioactive materials from the facility during LBEs other than DBAs, the reliability and performance of plant SSCs and personnel, and the effectiveness of programmatic controls.

(c) The safety analysis may not rely upon a single engineered design feature, human action, or programmatic control, no matter how robust, to address the range of LBEs other than DBAs.

Subpart B

Technology-inclusive safety requirements

§ 53.260 Normal operations.

Holders of licenses to operate commercial nuclear plants under this part must control public doses and dose rates in unrestricted areas from normal plant operations to meet the requirements in 10 CFR part 20.

§ 53.270 Protection of plant workers.

Holders of licenses to operate commercial nuclear plants under this part must control occupational doses to meet the requirements in 10 CFR part 20.

SRM-RELATED DELTA

Subpart B

Technology-inclusive safety requirements

Requirements related to radiation protection programs	
53.260	OL/COL holders meet 10 CFR part 20 (public doses)
53.270	OL/COL holders meet 10 CFR part 20 (plant workers)
53.425	• Define design features and functional design criteria • ALARA design objective of 10 mrem TEDE annual dose
53.430	Define design features and functional design criteria
53.450(g)(3)	Analysis of expected releases and doses to the public
53.850	Radiation protection program
53.1645	Reports of radiation exposure to the public
53.1239(a) (DC) 53.1209(b) (SDA) 53.1279(a) (ML) 53.1309(a) (CP)	• Design features supporting normal operations • How programmatic controls support meeting requirements • Design features supporting the protection of plant workers • How programmatic controls support meeting requirements
53.1369 (OL) 53.1416(a) (COL)	• Design features supporting normal operations • Radiation protection program • Design features supporting the protection of plant workers • Radiation protection program

SRM-RELATED DELTA

Subpart C

Design and analysis requirements

§ 53.400 Design features for licensing-basis events.

§ 53.410 Functional design criteria for design-basis accidents.

§ 53.415 Protection against external hazards.

§ 53.420 Functional design criteria for licensing-basis events other than design-basis accidents.

§ 53.425 Design features and functional design criteria for normal operations.

§ 53.430 Design features and functional design criteria for protection of plant workers.

§ 53.440 Design requirements.

§ 53.450 Analysis requirements.

§ 53.460 Safety categorization and special treatments.

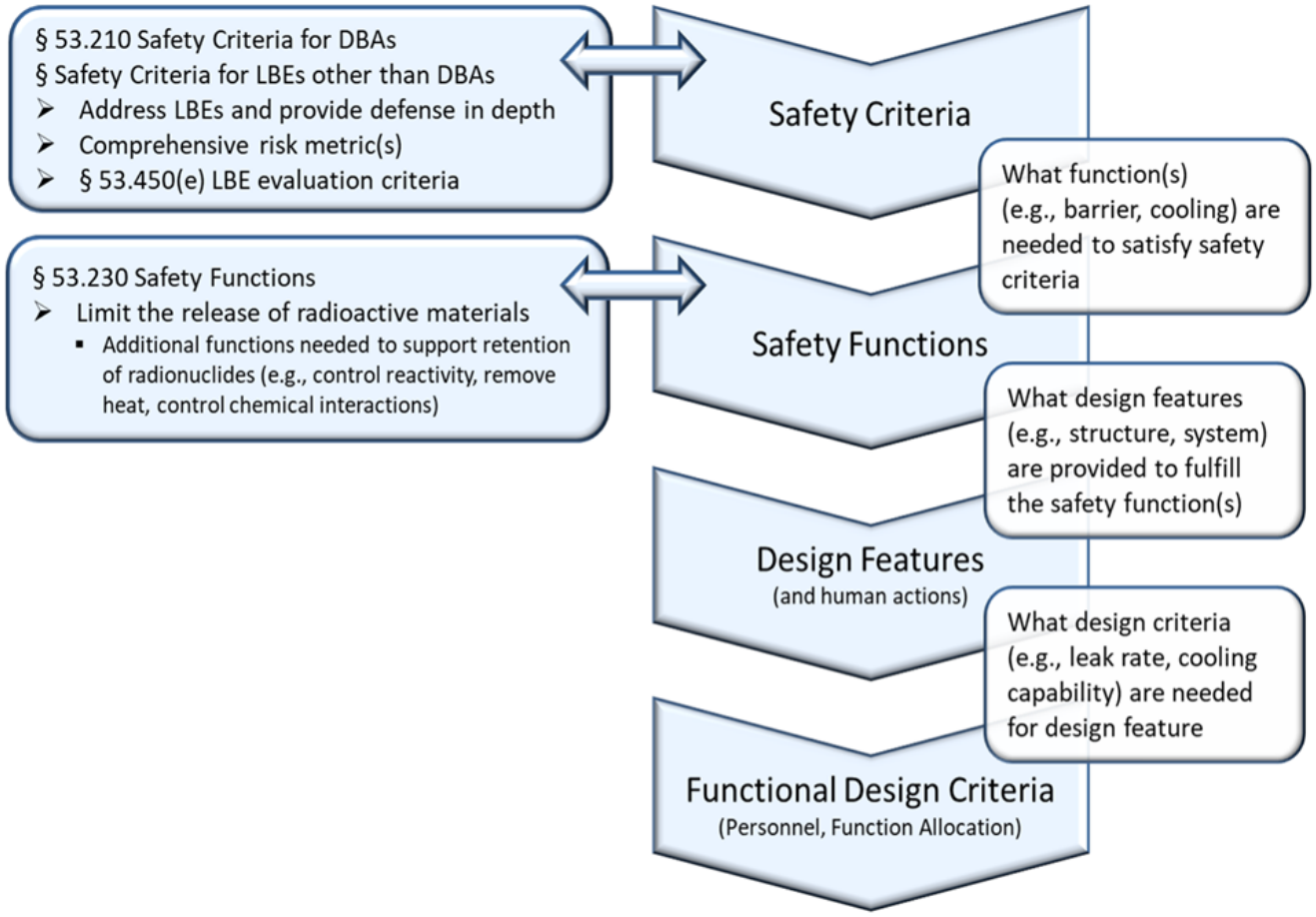
§ 53.470 Maintaining analytical safety margins used to justify operational flexibilities.

§ 53.480 Earthquake engineering.

Subpart C

Design and analysis requirements

Part 53 Hierarchy



Subpart C

Design and analysis requirements

§ 53.400 Design features for licensing-basis events.

- Design features must be provided such that, when combined with corresponding human actions and programmatic controls, the plant will satisfy the safety criteria and ensure that safety functions are fulfilled during LBEs.

§ 53.410 Functional design criteria for design-basis accidents.

§ 53.415 Protection against external hazards.

- Safety-related (SR) SSCs must be protected against or must be designed to withstand the effects of external hazards up to the design-basis external hazard levels.

§ 53.420 Functional design criteria for licensing-basis events other than design-basis accidents.

Subpart C

Design and analysis requirements

§ 53.440 Design requirements.	
(a)	• Demonstrate functional design criteria via analysis, test, etc.; • Evaluate operating, design and construction experience
(b)	Consensus codes and standards acceptable to NRC
(c)	Materials qualified for conditions
(d)	Evaluate possible degradation mechanisms
(e)	Design and locate to minimize probability and effects of fires and explosions
(f)	Consider safety and security together during design process
(g)	Subcritical condition during normal operations and after LBE
(h)	Long-term cooling during normal operations and after LBE
(i)	Design, analysis, staffing and programs cover all units, inventories
(j)	Physical barrier(s) maintained assuming aircraft impact
(k)	Control risk from chemical hazards of licensed material
(l)	Minimize contamination to facilitate eventual decommissioning
(m)	Criticality monitoring (alternative to § 70.24)
(n)	Consider human factors, functional analysis and function allocation

SRM-RELATED DELTA

Subpart C

Design and analysis requirements

§ 53.450 Analysis requirements.	
(a)	Requirement to have a probabilistic risk assessment (PRA)
(b)	Specific uses of analyses using PRA in combination with other generally accepted approaches for systematically evaluating engineered systems (LBEs, classification, defense in depth)
(c)	Maintenance and upgrade of analyses
(d)	Qualification of analytical codes.
(e)	Analyses of licensing-basis events other than design-basis accidents. • Evaluation criteria for each event or specific categories of LBEs • Means to identify event sequences significant for controlling risks
(f)	Analysis of design-basis accidents. • Deterministic methods from initiation to a safe stable end state
(g)	Other required analyses. • Fire protection • Aircraft impact • Doses to members of the public

Subpart C

Design and analysis requirements

PRA Acceptability

- Development, use, and maintenance of a PRA would be a key component in the proposed analysis requirements.
- The PRA, together with other techniques, would have required uses such as –
    - identify and categorize LBEs,
    - classify SSCs, and
    - evaluate defense in depth.
- Consistent with the current state of practice, acceptability of a PRA would be assessed based on the required uses of the PRA and the needs and scope of the application.
    - Consensus PRA standards would not be applied as a strict checklist of requirements for PRA acceptability determinations under the Part 53 proposed rule.
- NRC guidance on non-LWR PRA acceptability is currently available, which includes NRC-endorsed processes on the use of consensus PRA standards and PRA peer review.

SRM-RELATED DELTA/CLARIFICATION

Subpart D

Siting requirements

§ 53.500 General siting and siting assessment.

§ 53.510 External hazards.

§ 53.520 Site characteristics.

§ 53.530 Population-related considerations.

§ 53.540 Siting interfaces.

Subpart E

Construction and manufacturing requirements

§ 53.600 Construction and manufacturing – scope and purpose.

§ 53.605 Reporting of defects and noncompliance.

§ 53.610 Construction.

§ 53.620 Manufacturing.

Subpart E

Construction and manufacturing requirements

§ 53.620 Manufacturing.

- Management and control
  - Provides programmatic and organizational requirements
  - Supports compliance with the design and analysis requirements in subpart C
- Manufacturing activities
  - ML holder has the authority to establish controls at facility(s)
  - Manufacturing processes must be performed in accordance with the ML and the referenced codes and standards
  - A post-manufacturing inspection and acceptance process
- Control of radioactive materials
- *Fuel loading*
- Transportation
- Acceptance and installation at final place of operation

SRM-RELATED DELTA*

** Delta between ACRS Review and SECY-23-0021*

Subpart E

Fuel loading for manufactured reactor module

§ 53.620(d) *Fuel loading*.

- A manufacturing license may include authorizing the loading of fresh (unirradiated) fuel into a manufactured reactor under Part 70
- Specifies required protections to prevent criticality
  - At least two independent physical mechanisms in place, each of which is sufficient to prevent criticality assuming optimum neutron moderation and neutron reflection conditions
- Commission finding that a manufactured reactor module in required configuration is not in operation

SRM-RELATED DELTA

Subpart E

Fuel loading for manufactured reactor module

§ 53.620(d) *Fuel loading*.

- Holders of these Part 70 licenses must comply with the requirements of Subpart H to Part 70
- Procedures, equipment, and personnel required by the Part 70 license must be in place before the receipt of SNM at the manufacturing facility
- The loading or unloading of fresh fuel into or from a manufactured reactor and any changes to the configuration of reactivity control and prevention systems for the fueled manufactured reactor must be performed by a certified fuel handler meeting the requirements in Subpart F of this part
- For a manufactured reactor that is to be loaded with fresh fuel before transport to the place of operation, the ML must specify that transportation will be in accordance with Parts 71 and 73 of this chapter
- Security requirements

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Factory Testing of Fueled Manufactured Reactors

- Proposed § 53.620(d) would allow and establish requirements for the loading of fuel into a manufactured reactor at the manufacturing facility for transport to a site with a combined license

> Staff Requirements Memorandum (SRM)-SECY-23-0021
>
> 8. The staff should include factory fuel load provisions in the proposed rule. The staff should work with stakeholders following publication of the proposed rule to develop regulatory text that would also allow a holder of a manufacturing license to accomplish operational testing on a fueled manufactured reactor at the factory prior to delivery to the site where it will ultimately be used.

- Included question in *Federal Register* Notice
- Prepared and released preliminary draft material (i.e., not complete NRC management or legal review) to support discussions
  - ADAMS Accession No. ML24344A037
- Public meeting – January 8, 2025
- Consideration of comments received

SRM-RELATED DELTA

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Question in *Federal Register* Notice

- Should Part 53 include provisions for the testing of fueled manufactured reactors in the manufacturing facility?
  - What would be both practical and safe?
  - What tests are expected to collect data on fuel or other structures, systems, and components (SSCs)?
- What would be appropriate limits on operations?
  - Power levels
  - Durations (limit creation of byproduct material)
- What requirements could be revised given limitations on operation?
  - Licensing basis events, aircraft impact assessments, external hazards (seismic)

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Question in *Federal Register* Notice

- What regulations would be appropriate for manufacturing facility?
  - Construction (proposed § 53.610)
  - Operations (proposed §§ 53.710 and 53.715)
  - Personnel (proposed § 53.730)
- What licensing mechanism(s) should be considered for in-factory testing of manufactured reactors?
  - License for each manufactured reactor
  - License for manufacturing facility/multiple manufactured reactors
  - Inspections, tests, analyses, and acceptance criteria (ITAAC)

Subpart F

Requirements for operation

§ 53.700 Operational objectives.

***

(1) Each holder of an OL or COL under this part must maintain the **capabilities, availability, and reliability of plant SSCs** to ensure that the safety functions identified in § 53.230 will be performed if called upon during licensing-basis events (LBEs).

(2) Each holder of an OL or COL under this part must ensure that **plant personnel have adequate knowledge and skills** to perform their assigned duties that support the performance of the safety functions identified in § 53.230.

(3) Each holder of an OL or COL under this part must implement **plant programs** sufficient to ensure that the safety functions identified in § 53.230 will be performed if called upon during normal operations and LBEs.

Subpart F Organization of Sections	
§ 53.710 - § 53.720	SSCs
§ 53.725 - § 53.830	Personnel
§ 53.845 - § 53.910	Programs

Subpart F

Requirements for operation

§§ 53.725 - 53.830: General staffing, training, personnel qualifications, and human factors requirements

Sections 53.725 – 53.830 include the following key areas:

- Content of application requirements (§ 53.730)
  - Human factors engineering (HFE) has a safety function focus (versus generic application to a control room)
  - Facility-specific staffing plans and “engineering expertise”
- Conditions of license for facility licensees (§ 53.740)
  - Allows for automatic load following
  - Addresses online refueling oversight
- Operator licensing requirements for specifically-licensed Senior Reactor Operators (SRO) and Reactor Operators (ROs) (§§ 53.760-53.795)
  - Addresses use of customized operator licensing programs
  - Allows facility licensees to administer license exams
- Requirements for Generally Licensed Reactor Operators (GLROs) (§§ 53.800- 53.820)
  - Establishes criteria for “self-reliant-mitigation” facilities.
  - Contains the general license for GLROs
- Plant staff training requirements (§ 53.830)

Subpart F

Requirements for operation

Other highlights from §§ 53.725 - 53.830

- Load following is permitted under 53.740 if one of the following is immediately capable of refusing unsafe demands:
  1) an automatic protection system that utilizes setpoints more conservative than those otherwise credited for the purposes of reactor protection; or
  2) an automated control system; or
  3) intervention of an RO, SRO, or GLRO.
- Prescriptive timeframes for establishing training programs are no longer used; 53.830 requirement is based on needs.
- Staffing plans are proposed by applicants under 53.730, with HFE analyses and performance-based tests being used to determine operator numbers, qualifications, and locations (approved plan then becomes license condition).
  - A flexible requirement for engineering expertise is used in lieu of traditional Shift Technical Advisor staffing.
  - A location-neutral approach is taken to operator staffing; for example, control room staffing is not prescribed.
  - “Self-reliant-mitigation” facilities do not require these HFE-based staffing analyses and, instead, only have GLRO oversight and engineering expertise requirements.

Subpart F

Requirements for operation

Self-reliant-mitigation facilities and GLROs

- § 53.800 – Criteria for self-reliant-mitigation facilities
  - No human actions to meet radiological consequence criteria, address LBEs, or provide for adequate DID
  - Safety functions not allocated to human action
  - Reliance upon robust and highly reliable safety features
- § 53.805 – Facility licensee requirements for GLROs
  - Facilities must continue to meet the criteria of 53.800 (failure would be a reportable unanalyzed condition)
- § 53.810 – General license for GLROs
  - Grants similar level of administrative authority as an SRO
  - No application needs to be submitted for GLRO licensing
  - Individuals operating under license subject to conditions
  - License can still be suspended on an individual basis
- § 53.815 – GLRO training, exams, & proficiency
  - SAT-based training program is required
  - Uses customized, Commission-approved exam programs
  - After approval, GLRO programs are facility-administered
  - Facilities determine requalification exam periodicity
  - Simulation facilities do not require Commission-approval
- § 53.820 – Cessation of individual applicability

Subparts G, H, I, J, and M

- Subpart G — Decommissioning Requirements
- Subpart H – Licenses, Certifications, and Approvals
- Subpart I — Maintaining and Revising Licensing-Basis Information
- Subpart J — Reporting and Other Administrative Requirements
- Subpart M — Enforcement

Wrap Up Discussion and Questions

Acronyms

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
AEA	Atomic Energy Act of 1954
AERI	Alternative Evaluation for Risk Insights
ALARA	as low as is reasonably achievable
ARCAP	Advanced Reactor Content of Application Project
CFR	*Code of Federal Regulations*
COL	combined license
COL-TMR	combined license for testing of manufactured reactors
CP	construction permit
DBA	design-basis accident
DC	design certification
DID	defense in depth
EP	emergency planning
FR	*Federal Register*
FRN	*Federal Register* Notice
GLRO	generally licensed reactor operator
GEIS	Generic Environmental Impact Statement
HFE	human factors engineering
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
LB	licensing basis
LBE	licensing-basis event
LWR	light-water reactor
ML	manufacturing license
mrem	millirem
NEIMA	Nuclear Energy Innovation and Modernization Act
NRC	U.S. Nuclear Regulatory Commission

Acronyms

NRR	Office of Nuclear Reactor Regulation
NUREG	U.S. Nuclear Regulatory Commission technical report designation
OL	operating license
OMB	Office of Management and Budget
PRA	probabilistic risk assessment
QA	quality assurance
QHO	quantitative health objectives
RG	Regulatory Guide
RO	Reactor Operator
SAR	safety analysis report
SAT	systems approach to training
SDA	standard design approval
SECY	Office of the Secretary
SNM	special nuclear material
SR	safety-related
SRM	Staff Requirements Memorandum
SRO	Senior Reactor Operator
SSC	structure, system, or component
TICAP	Technology-Inclusive Content of Application Project
TEDE	total effective dose equivalent
TS	technical specifications

Backup Slides

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White Paper (ML24344A037)

- Provided to support discussions
- Should not be interpreted as official agency positions

- White Paper organized to provide:
  - Description
  - Draft preliminary rule text (§ 53.1480)
    - Combined license for testing manufactured reactors (COL-TMR)
    - Commission findings on operating states*

* See also FRN Question 7. under Part 53, Subparts E and H—Manufacturing Licenses

7. Some stakeholders have suggested that a fueled manufactured reactor with appropriate protections against criticality should not be categorized as a utilization facility under NRC regulations or Section 11cc. of the AEA.

The NRC is seeking comment on possible approaches where the NRC could find that a fueled manufactured reactor would not be a utilization facility, the basis for such a finding, and the potential benefits of and potential issues with such a finding.

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White Paper

- Provided to support discussions
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- White Paper basic approach
  - Building from proposed § 53.620(d)
    - Unirradiated fuel loaded (manufacturing license; Part 70)
  - Limit introduction of byproduct material
    - Radioactive inventory, decay heat
    - Assume in-factory conditions for licensing-basis events
    - Limited consequences assumed in categorizing hazards
  - Consideration of various regulations and licenses
    - Part 53 (Manufacturing license, combined license)
    - Part 70 (Special nuclear material)
    - Part 30 (Byproduct material)
    - Parts 71, 73, 74 and others as needed

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White Paper

- Provided to support discussions
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- Selected White Paper examples (technical requirements)
  - Limit power level (≤ 5% rated thermal power (commercial))
  - Limit inventory (indirectly via defining restrictive safety criteria (Part 20 annual dose))
  - Licensing-basis events
    - Identified for reactor as tested (e.g., fresh fuel)
    - Mitigated without reliance on human actions
      - Consistent with use of generally licensed reactor operators (GLROs)
    - Design features of manufacturing facility and manufactured reactor
  - Holder of manufacturing license ensures testing does not adversely affect downstream activities (storage, transport, deployment)

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White Paper

- Provided to support discussions
- Should not be interpreted as official agency positions

- Selected White Paper examples (technical requirements)
  - Possible alternatives mentioned in draft paper:
    - § 53.440(j) (*aircraft impact*) would not apply
    - §§ 53.415, 53.480, and 53.510 (*external hazards*) would not apply
      - Based on limited consequences, commercial codes
    - § 53.610 (*construction*) would apply to portions of manufacturing facility
    - §§ 53.710 and 53.715 (*SSC configuration control*) would apply for testing
    - §§ 53.730(a) through (e) (*human factors*) would apply
    - § 53.730(f) (*staffing plan*) would be supplemented
      - Test Engineer, Reactor Engineer, GLRO
    - §§ 53.870 and 53.880 (*ISI/IST, Integrity assessment)* would not apply
    - Alternate decommissioning funding requirements (such as Parts 70 and 30) might apply

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White Paper

- Provided to support discussions
- Should not be interpreted as official agency positions

- Selected White Paper examples (licensing construct)
  - COL-TMR
    - Applicable to portions of manufacturing facility and each manufactured reactor (1 through n)

	Testing criteria for first reactor	Testing criteria for subsequent reactors	Criteria for final place of operation
Manufacturing facility	ITAAC (COL-TMR)	§§ 53.710 and 53.715	n/a
Manufactured reactor	ITAAC (COL-TMR (incl ML))	ITAAC (COL-TMR (incl ML))	ITAAC (COL (incl ML))

    - Updates to the ITAAC schedule under § 53.1449(a) and ITAAC closure notifications under § 53.1449(c) may address multiple manufactured reactors that are under fabrication or planned to be fabricated under the ML and tested under the COL-TMR
    - Conforming changes (e.g., § 53.620(d))