Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: 735th Meeting Advisory Committee On Reactor Safeguards

Docket Number: N/A

Location: Rockville, Maryland

Date: May 6, 2026

Work Order No.: NRC-0644 Pages 1-248

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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735TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

WEDNESDAY

MAY 6, 2026

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

COMMITTEE MEMBERS:

GREGORY H. HALNON, Chair

DAVID A. PETTI, Vice Chair

CRAIG D. HARRINGTON, Member-at-Large

ANNIE M. KAMMERER

WALTER L. KIRCHNER

ROBERT P. MARTIN

SCOTT P. PALMTAG

THOMAS E. ROBERTS

MATTHEW W. SUNSERI

ACRS CONSULTANTS:

VICKI M. BIER

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

ALSO PRESENT:

CHRISTIAN ARAGUAS, NRC

MIKE BALAZIK, NRC

LARRY BURKHART, NRC

AMY D'AGOSTINO, NRC

ELIJAH DICKSON, NRC

WILLIAM KENNEDY, NRC

EDWIN LYMAN, Union of Concerned Scientists

ADAM STEIN, The Breakthrough Institute

ANTHONY ULSES, NRC

JUSTIN VAZQUEZ, NRC

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

8:30 a.m.

CHAIR HALNON: Good morning. This meeting will now come to order. This is the first day of the 735th meeting of the Advisory Committee on Reactor Safeguards. I am Greg Halnon, Chairman of the Committee.

ACRS members in attendance in person are Craig Harrington, Annie Kammerer, Walt Kirchner, Robert Martin, Scott Palmtag, Dave Petti, and Tom Roberts. Member Matt Sunseri is attending virtually. Our consultant Vicki Bier is in attendance. Ron Ballinger is in the building, but he has got a task he is working on and he may poke his head in during the meeting.

If I have missed anyone, either ACRS members or consultants, please speak up.

Weidong Wang of the ACRS staff is the Designated Federal Officer for this Committee meeting.

We have a quorum for today's meeting.

The ACRS was established by the Atomic Energy Act and is governed by the Federal Advisory Committee Act. Under the Atomic Energy Act, ACRS must advise the Nuclear Regulatory Commission on the hazards of proposed and existing reactor facilities

and the adequacy of proposed safety standards.

Following Executive Order 14300, the Committee focuses on those activities necessary to fulfill its statutory obligations. As a result, the ACRS is prioritizing the review and reporting of new reactor facilities and new or revised proposed safety standards with a particular attention to issues that are unique, novel, and noteworthy. The Committee will consider other nuclear safety matters at the direction of the Commission.

Please note the ACRS speaks only through its published Letter Reports. All member comments should be regarded as only the individual opinion of that member, not a Committee position.

Information about ACRS activities, such as letters, meeting rules, and transcripts are on the NRC public website and could be found by searching "About Us ACRS" on the NRC's home page.

The ACRS provides an opportunity for public input and comment on relevant technical issues during our proceedings. For this full Committee meeting, we have received no written statements or requests for verbal comment. Written statements may be forwarded to today's Designated Federal Officer.

We have set aside time in this meeting for

public verbal comments on the Committee technical matters.

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

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

The Teams chat feature is for only communicating IT issues or brief meeting logistics. Please do not use it for comments or questions on the topics under Committee discussion.

For everyone in the room, please put all your electronic devices in silent mode and mute your laptop microphone and speakers. In addition, please keep sidebar discussions in the room to a minimum, since the ceiling microphones are live.

For the presenters, your table microphones are unidirectional and you will need to speak into the front of the microphone to be heard online.

Finally, if you have any feedback for the ACRS about today's meeting, please fill out the public

meeting feedback form on the NRC's website.

Our first item for this full Committee meeting is the Proposed Rule and Licensing Requirements for Microreactors and Other Reactors of Comparable Risk Profiles, also known as Part 57.

And before I turn it over to Dave Petti, the Subcommittee Chair, to lead the discussion, I wanted to highlight a few important points.

First of all, today's agenda is very general. We made it very flexible, so that most of our time will be spent in Committee deliberation on the rule.

We will begin with a short presentation from the NRC staff, followed by an opportunity for public comments to the Committee.

The public comment portion will be limited to one and a half hours, so the Committee can proceed with letter writing. Individual commenters will be asked to limit remarks to three minutes.

If time remains, we can double back to those that made comments and wish to expand on their comments, but we want to give everybody an opportunity -- anticipating, potentially, a lot of comments.

Secondly, I wanted to clarify that any public comments that are made in this meeting are

solely for the benefit of ACRS deliberations. They will not be considered as part of the NRC docketed rule on www.regulations.gov. If you do want to have your comment considered in the rulemaking, then you have to get into the regulations.gov website.

Third, if you wish to make a public statement to the Committee, secure your place in line -- again, anticipating many comments. Please let us know by emailing Larry Burkhart at lawrence.burkhart -- that's B-U-R-K-H-A-R-T -- @nrc.gov and weidong.wang -- that's W-E-I-D-O-N-G, dot, W-A-N-G -- @nrc.gov, so we can create a list of speakers and get a line-up. Larry's and Weidong's email addresses are also posted in the Teams chat.

After we've gone through the speaker list, we'll take additional comments, as I mentioned, if time permits.

Finally, on May 20th, the NRC staff will hold a public meeting regarding this proposed rule, and details on that meeting will be on the NRC website.

So, for the members, as we go through the presentation this morning, there's going to be sections where we just need to be quiet and listen, and let Elijah or the presenter say, "Now is the time

to ask questions." So, that will be efficient and allow them to get through their notes, so through their talking points, and then, we can ask the questions. So, let's try to limit, if not eliminate, interruptions until they are ready for questions.

Any opening comments by members?

(No response.)

CHAIR HALNON: Okay. Then, with that, Dave, I'll turn it over to you to lead us through this morning's discussion.

VICE CHAIR PETTI: Thank you. We're going to hear today from a member of the staff on Part 57 licensing requirements for microreactors and other reactors. There's a fair number of slides here. So basically, our agenda, we dedicated most of the morning to hear from the staff.

A couple of comments. To hold to the schedule, we won't get too (audio interference). But if there's fewer public comments, we might be able to (audio interference).

With that, let me ask Christian to start.

MR. ARAGUAS: Well, good morning, Mr. Chairman, Members of the Committee, and thank you again for the opportunity to be here today with you.

My name is Christian Araguas and I serve

as the Director for our Division of Engineering, Office of Nuclear Regulatory Research. And I serve as one of two executive sponsors for development of the rule, alongside with Cinthya Roman, who's in the back of the room there. She is currently serving as the Deputy Director in our Division of Fuel Management in the Office of Nuclear Material Safety and Safeguards.

So, before I dive or we dive into the presentation, I did want to just to acknowledge the staff across the Agency that have worked to support this rule and bring this framework forward. You know, they have put in a tremendous amount of effort to bring this rule forward. And developing a new licensing construct of this breadth on an accelerated timeline has required a serious technical, legal, and regulatory effort. And I appreciate the discipline with which the staff has approached this.

So, this rulemaking comes at a time when the national conversation around energy security, energy reliability, and advanced nuclear technologies are moving quickly.

Congress and the Administration have made it clear that the NRC must be prepared to support timely licensing of these new technologies. At the same time, the Agency's response has to be more than

simply moving faster. It has to be making sure we're applying the regulatory process in a way that is appropriately matched to the designs we are reviewing, while preserving the technical rigor and independence that underpin public confidence in the NRC. That is what Part 57 is intended to do.

This is not a replacement for our existing regulatory frameworks, and it is not an effort to lower the safety bar. It's a targeted framework for a narrower class of microreactors or comparable designs where the passive safety features, standardization, and lower potential offsite consequences may justify a more front-loaded and more repeatable licensing approach. But any framework built on those judgments deserves careful technical scrutiny.

The Committee's role here is important because this framework introduces a number of new approaches and technical considerations for the Agency -- whether the eligibility criteria are properly calibrated; whether the balance between flexibility and regulatory certainty is the right one, and whether the Agency has appropriately aligned licensing efficiency with adequate assurance of protection. We look forward to your advice on these issues today and

the discussion this morning.

And with that, I turn it over to staff. Thank you.

MR. DICKSON: Okay. With that, we can go ahead and move on to the --

VICE CHAIR PETTI: Elijah, your microphone.

CHAIR HALNON: Bring it right up to you.

MR. DICKSON: With that, we can go ahead and start with the agenda. So, here's today's agenda.

I will begin with the purpose of the Proposed Part 57 Rulemaking and discuss its overarching principles that guided its development.

We'll then walk through the structure of the rule, including key subparts that the ACRS may be interested in reviewing, starting with 57.25, which is entitled, "Applicability." This is found in Subpart B of the eligibility section of the rule, which defines which reactors are eligible for the Part 57 framework.

From there, we'll cover the Part 57 Licensing Pathways.

CHAIR HALNON: Elijah?

MR. DICKSON: Yes?

CHAIR HALNON: I'm going to put you on

hold just for a second. There's some static on the line.

(Whereupon, at 8:40 a.m., the foregoing matter went off the record and went back on the record at 8:42 a.m.)

CHAIR HALNON: Okay, Elijah, go ahead.

MR. DICKSON: Okay. How does this sound? It's also sounding okay on my end?

CHAIR HALNON: It's good.

MR. DICKSON: Okay. All right.

So, I'll go back to discussing the Licensing Pathways of Part 57. From there, we'll follow the Implementation Guidance that we developed. It's called NUREG-2271 and discusses, I think, some of the pertinent chapters that the members would be interested in following or reviewing.

We'll then finally close with the schedule for the rulemaking process.

This structure is intended to give the Committee a clear picture of both the regulatory approach and practical deployment and implementations of the Part 57 framework.

So, let's go ahead and go on to the purpose.

This slide lays out the motivation behind

the proposed Part 57 framework. The core issues that are current or existing licensing frameworks: Part 50 and Part 52, Part 53 to an extent, and how we regulate research and test reactors that are not designed for high volume, rapid deployment of microreactors and other reactors of similar or comparable risk profiles.

Part 57 is intended to provide a more streamlined, scalable regulatory pathway. Part 57 does this by enabling applicants to submit a joint application for Construction Permit and Operating License, which allows the NRC to complete safety, security, and environmental reviews at one time, and also, have a single opportunity for a public hearing and ACRS reviews. This reduces repetition, shortens review timelines, and supports standardized deployment models.

The key concept that we are utilizing in this framework is a plant and site parameter envelope, which is a technology-neutral approach borrowed from other regulatory frameworks to streamline subsequent applications to ensure consistent assumptions across a wide variety of reactor types and potential sites.

So, in practice, after a first-of-a-kind regulatory review has been approved and completed, nth-of-a-kind approvals, but mostly focused on site

reviews to evaluate whether the site fits the originally approved plant parameter envelope, and you can leverage generic data for siting purposes.

Any questions?

(No response.)

All right. So, this slide is going to talk about providing some context. It summarizes the primary foundations guiding the development of Part 57. They reflect statutory mandates, executive direction, and longstanding Commission policies on risk-informed and performance-based regulations.

First, of course, we have the Atomic Energy Act of 1954, as amended. We often went back to the AEA in crafting several areas of Part 57.

There are also two primary, very recent mandates that are driving NRC's transformation and two acts from Congress. This would be NEIMA and the ADVANCE Act, as well as three Executive Orders that were issued last year.

NEIMA establishes the foundation for technology-inclusive, risk-informed, and performance-based regulatory frameworks; most notably, the development of Part 53.

The ADVANCE Act accelerates this shift by emphasizing regulatory efficiencies and more proactive

NRC posture in supporting the deployment of innovative nuclear technologies.

Together, NEIMA and the ADVANCE Act define both what must change and how quickly it must change.

The Executive Orders also provide operational urgency.

Executive Order 14300 directs the reform of the NRC itself to improve agility and responsiveness.

Executive Order 14156 establishes a National Energy Emergency, underscoring the need for reliable and diversified energy supplies.

Executive Order 14154 reinforces the national priority to expand affordable, reliable American energy.

The cumulative effects of the acts and the Executive Orders make a clear mandate to modernize regulatory frameworks, streamline mission execution, and support national energy security strategies.

But we would also like to talk about some of the kind of like foundational policies that we considered and looked at extensively when developing the framework.

We draw heavily on Commission Policy SRM-SECY-98-144 -- it's a white paper on developing

risk-informed and performance-based regulations; this is a Commission expectation -- as well as the Commission's Policy Statement on Advanced Reactors. There are also more recent Staff Requirement Memoranda in regards to "Microreactor Licensing and Deployment Considerations for Fuel Loading and Operational Testing," and there's SECY-25-0052 for "Nth-of-a-Kind Microreactor Licensing and Deployment Considerations."

So, with that, we can go ahead and move on to discussing the slide on slide 5. Thank you. This is the rule structure.

On this slide, we are previewing how Part 57 is organized. The intent is to make it a one-stop-shop type of regulatory framework covering both conception of initial designs, thoughts, all the way to deployment, as well as decommissioning. It's all here. There are areas of the rule that kick out to other frameworks: Part 70, Part 71 for other items. But for the most part, we like to think of it as a one-stop-shop type of rule.

So, the structure deliberately focuses on technical requirements through Part 50 concepts, while drawing upon licensing approval mechanisms under Part 50 and Part 52. And this is to keep the reviews efficient and predictable.

Subpart B is entitled the "Eligibility." This is effectively the entry gate into the rule. The first entry gate is 1 rem Total Effective Dose Equivalent criterion and the second one is a fuel mass limit, plus the use of six design criteria attributes that characterize microreactors and other reactors of comparable risk profiles that we're trying to attract into the Part 57 framework.

We have Subpart C. This is the Construction Permits and Operating License. This framework leverages the joint application of a CPOL model, front-loading design and operational programs, as well as streamlining ACRS reviews and enable generic finality, where appropriate. I'll talk a little bit about these in future slides.

We have the Subpart D, "Manufacturing License." It authorizes standard manufacturing and affords finality for designs that downstream applicants can reference. It's a key aspect of scalable deployment.

We have Subpart E for "Standard Design Approvals." This allows full designs or major portions thereof to be reviewed by the staff. This is very similar to Part 52's Subpart E, "Standard Design Approval," which is a staff-level approval of a

reactor design. We'll have a slide on that later.

We have Subpart G in regards to the "Irradiated Fuel Storage, Decommissioning, and License Termination Requirements."

Subpart I, "Transportation Package Design Certification." Here, we are leveraging Part 71 to accommodate reactors that are either to be considered a package or the contents of for the purposes of transportation.

And we also have Subpart P, "Operator Licensing and Human Factors."

And then, lastly, "Fitness for Duty" under Part 26, Subpart P.

A lot of different aspects of the rule.

Any questions?

(No response.)

Okay. So --

CHAIR HALNON: This is Greg. I was just trying to think. Wasn't there a quality appendix as well?

MR. DICKSON: The quality assurance requirements would be in Subpart C, and there's a high-level requirement for quality assurance programs there.

CHAIR HALNON: Okay.

MR. DICKSON: So, Subpart C -- I'll state it here -- is very similar to how 50.34 is set up for the contents and applications of an SR. And so, all of those technical requirements in Part 50, it's structured -- well, we didn't copy them over, but they're structured that way. So, in an application, you would see, like, the quality assurance program and different -- within their submittal it would be presented there.

CHAIR HALNON: Okay. But there's no equivalent to, like, Appendix B, which we're all used to?

MR. DICKSON: No, there's not.

CHAIR HALNON: Okay. Thanks.

MR. DICKSON: Okay. So, we'll go ahead and start on slide 6. And I suspect this is where we're probably going to get a lot of comments, is Subpart B.

Now, the NRC based the development of the proposed Part 57 framework on existing licensing practices for non-power and other utilization facilities that, by design and operational characteristics, present low risks of radiological consequences. These characteristics have designers approach safety by emphasizing accident prevention

with inherent self-limiting reactivity feedback mechanisms and passive safety systems for heat and decay heat removal without the reliance of complex active systems.

The NRC uses these characteristics to support a set of requirements to determine which applicants would be eligible for the proposed Part 57 framework. And these are located in 57.25, entitled, "Applicability," and 57.30, "Design Criteria Attributes." The criteria does not include a power-level equivalent.

And the purpose is also to avoid licensing complications with more complex designs having this eligibility or this Subpart B eligibility criteria. As we know, more complex designs do achieve economies of scale, but there are tradeoffs. You need active safety systems, different quality requirements, and operational programs to run these machines. Therefore, it would make it more difficult to review a very large machine that needs active safety systems in the type of timelines we're looking at in meeting the Executive Order for rapid licensing of microreactors.

So, as such, the staff had proposed an attribute-based approach to encourage innovation of

advanced reactors. There are reactors that have safety profiles, that they achieve higher power levels than others with the same amount of special nuclear material or less. So, we didn't want to limit the rule to a very specific power level.

So, with that, we can take some questions or we can move on to slide 7, where we start talking about each of these --

MEMBER MARTIN: Elijah?

MR. DICKSON: Yes.

MEMBER MARTIN: This is Bob Martin.

MR. DICKSON: Uh-hum.

MEMBER MARTIN: In establishing this eligibility, do tabletop, say, some of the designs that are out there --

MR. DICKSON: We looked at quite a few, yes. The Federal Register Notice, there's a rather long discussion of the staff's analysis of each of these, these three criterias.

So, the first one being the dose-based entry criteria. That is something that we have longstanding doing. The material limit of 10 metric tons uranium, we performed, you know, a survey that actually designs. We have a discussion as to how we got there. And then, also, you know, considered some

of the designs that were in-house, wondering whether or not they could be -- I guess that's probably the best tabletop we had. You know, what could be effectively -- what could effectively utilize the framework? So, there's a discussion of that analysis in the FRN.

MEMBER MARTIN: Can you share, say, what didn't make the list as being eligible? We're certainly looking at a lot of small reactors.

MR. DICKSON: Yeah.

MEMBER MARTIN: You know, we use the term "micro," small, and, of course, the large.

MR. DICKSON: We didn't call them out in the Federal Register.

MEMBER MARTIN: Right. Of course.

MR. DICKSON: But we have the 1 rem entry criteria, and that's scalable, right? So, effectively, any reactor could have used the rule. If you have a very large site boundary, any reactor can get inside of the rule.

MEMBER MARTIN: Right.

MR. DICKSON: And so, we knew there are reactors out there that we don't want in the Part 57 framework because that could hurt our ability to do rapid licensing and we could get bogged down with very

detailed analyses.

But there's some that are kind of on the cusp, and we'll have to leave it for the submittals, you know, pre-application engagement and the submittals to --

MEMBER MARTIN: Well, we might find that some designs that we are calling small modular reactors --

MR. DICKSON: Yeah.

MEMBER MARTIN: -- could --

MR. DICKSON: That's right, yeah.

MEMBER MARTIN: -- could meet the microreactor definition.

MR. DICKSON: Yup, I think so. Right. That would be a reactor of low consequence.

MEMBER MARTIN: Right.

VICE CHAIR PETTI: So, this is Dave.

I mean, some of this is really in the details. For instance, I will mention a design that no one is pursuing, but that fits: MHTGR, which the Commission looked at in the nineties --

MR. DICKSON: Yeah.

VICE CHAIR PETTI: -- meets 1 rem.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: Meets the heavy metal

loading.

MR. DICKSON: That's right.

VICE CHAIR PETTI: That's 350 to 450 megawatt thermal, depending on which design you look at. Yet, I don't see it as really fitting in with, let's call it, the tenor of the rule. It's not what I think we would classify as fitting here, because it's a very large reactor. I think the vessel is the size of a BWR vessel and it's only producing, you know, 250 to 450 megawatts.

So, that's why -- I mean, I understand that you want to do power level, but it's simple. The 1 rem is also, you know, it does require a certain sharp pencil, if you will.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: And coming in as an entry criteria, you have to be fairly well down the path of your analysis, if you think you're going to be on the edge.

MR. DICKSON: Right.

VICE CHAIR PETTI: You know, and you're not going to be able -- so, I think entry criteria should be simple, because they need to make this decision early in their thinking. If you're a micro, I don't think that's a problem, you know, a classic

micro, a 10-megawatt thermal, even a 50-megawatt thermal. It's the ones that are a little bit larger would look at this and go, oh, well, maybe we could do that. You really don't -- you need a sharper pencil to get there. I don't know if that's an intended -- I don't want to see an unintended consequence here.

MEMBER KIRCHNER: I echo Dave's concerns. I can design a lot of reactors 10 metric tons heavy metal.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: Hundreds of megawatts, probably even more. That's not what's initially driving towards a microreactor rule. And we've just now reviewed several concepts that probably would engage with you on the last bullet and suggest that they can meet your entry criteria.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: So, this subverts 53 and probably has the overall unintended consequence of having large SMRs, if not larger reactors --

CHAIR HALNON: So, Walt, this is Greg.

MEMBER KIRCHNER: -- testing you under this rule.

CHAIR HALNON: Isn't that the second part, though, the design criteria attributes? What you're

talking about is --

MR. DICKSON: Is our requirements.

CHAIR HALNON: -- based in the simplicity of it. So, you can make these larger, complex machines that may meet the entry criterion, but won't pass through the design criterion.

VICE CHAIR PETTI: I think they will.

CHAIR HALNON: They will in many cases?

VICE CHAIR PETTI: Yeah, I mean, those criteria, almost every gas --

CHAIR HALNON: I guess we need to hear them then. Are you going to --

VICE CHAIR PETTI: Actually, a small sodium system would fit that.

CHAIR HALNON: Yeah.

VICE CHAIR PETTI: Not, for instance, TerraPower. If you look at their highest dose base, they're just a little over that. So, they don't make it. But just make it a little bit smaller --

CHAIR HALNON: Yes.

VICE CHAIR PETTI: -- or you make the site a little bit bigger.

CHAIR HALNON: A little bit bigger.

MEMBER KIRCHNER: If they come in with an EBR-II clone --

VICE CHAIR PETTI: Oh, an EBR-II would easily --

MEMBER KIRCHNER: -- it would fit within this.

VICE CHAIR PETTI: Yes.

MEMBER KIRCHNER: And that's probably not the intent.

VICE CHAIR PETTI: It's right on the edge, because wasn't it 60-some megawatts?

MEMBER KIRCHNER: Yeah, it was in that category.

CHAIR HALNON: Well, let's go through the design criteria attributes and see how those things fall out. It may be that maybe my comment is moot, but --

VICE CHAIR PETTI: I have another broader question then. You don't have to answer now, but I need it.

I've been stepping back, because we know Part 53 -- at least I do -- fairly well. I look at Part 57, and from a strict licensing basis and establishing the licensing basis, how is it different than Part 53 per se? I know the differences with 15.52, because this is more risk-informed --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- and performance-based. And I do understand in the natural phenomena there are some important differences. I get you. But let's just talk about kind of the traditional licensing basis and internal events. As I read everything, they sound awfully similar. And I think that's one of the things the applicant is going to look at and go, okay, you know, how different is it?

And this is particularly when I read the Implementation Guidance.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: Because when you read the entry criteria, you're, like, okay, and at the high level, they look really different. But when you get to the guidance, it gets a lot fuzzier for me, as I read the words on the page. So, I think it would be worth us talking through that. That's kind of at the end. We could come back around to get some better clarify. Because maybe it's something that I just missed in reading the Implementation Guidance, because it's large, but I just didn't see as much differentiation as I thought there might be.

MR. DICKSON: Got it. Okay.

VICE CHAIR PETTI: Again, just something

to put on -- I've got it on my list here.

MR. DICKSON: Okay. I think we can speak to that in some of the later slides.

VICE CHAIR PETTI: Okay. Good.

MEMBER HARRINGTON: And this is Craig.

I had the same -- you can factor this into what you say next.

MR. DICKSON: Yeah.

MEMBER HARRINGTON: But to what extent is this entry criteria subjective, in the sense that, you know, like the discussion, an applicant would look at this and say, yeah, I think I would fit in that.

MR. DICKSON: Uh-hum.

MEMBER HARRINGTON: And then, they -- I get the sense -- basically have to come to the staff and get that answer.

MR. DICKSON: Yeah.

MEMBER HARRINGTON: So, you know, that in itself creates a bit of --

MEMBER KIRCHNER: Uncertainty.

MEMBER HARRINGTON: -- you know, another thing to step over to decide --

MR. DICKSON: Uh-hum.

MEMBER HARRINGTON: -- are you in or are you out, and how do you get there? And how does the

system define that across a diversity of designs.

MR. DICKSON: Yeah.

MEMBER HARRINGTON: So, you can kind of thing about that.

MR. DICKSON: Got it. I think we can talk about that through the rest of the other slides.

MEMBER KIRCHNER: Conceptually, Elijah, when we think of micro language, we're thinking of something that's factory manufactured, potentially loaded with fuel and transported. And that concept, and you -- you, the staff, collectively -- you mentioned earlier some of the SECYs and such -- have looked at this in great detail. Some of these other larger SMRs that would get in under the first two criteria aren't necessarily in that kind of concept of deployment.

I don't know how you would -- whether it would be even correct to say you have to be able to load fuel and transport it to qualify as a microreactor, but that's part of what you're trying to address, this idea of high-volume --

MR. DICKSON: Yes, sir.

MEMBER KIRCHNER: So again, I'll just amplify Dave's concern that -- I would say 50 megawatts is a good criteria. Ten metric tons, it's

going to cause the Agency some significant dialog with a lot of applicants and time and effort to say, yes or no, you're going to meet these attributes.

I'm just afraid of the unintended consequence of putting the Agency in a position where it's using all of its energies in discussion with applicants about whether I am really inherently safe or passively safe. I'm simplified enough; I don't have the severe accident potential, whatever. It's just a concern that I have for its implementation going forward.

MEMBER KAMMERER: So, this is Annie Kammerer.

To foot stomp that a little bit more, I work with a lot of microreactor designs and I'm familiar, pretty in-depth. And 10 metric tons is a very large reactor.

And I think the objective here is to make it a simpler approach for a manufacturer of reactors, not necessarily fueled onsite; fueled in the factory, but for that smaller category of reactor.

And I think that the unintended consequence may be that down the road you're wanting to do things that make the licensing pathway more efficient for those smaller reactors. But since you

set the bar at such a big reactor, that you have to consider the implications for something that big, when really it might limit you for those that truly are the smaller concept reactor that I think is the spirit of the rule.

Over. Sorry.

MEMBER ROBERTS: Yeah, this is Tom Roberts.

Not to play with this too much, but they made a point about the timing of when you can reliably determine a dose consequence like 1 rem. And in the draft rulemaking, that tied it to the maximum hypothetical accident, maximum credible accident concepts.

And it isn't clear to me how crisply you can define those early in the design when you're entering. So, that last bullet on this slide really ties to how you can measure the inherently safe and passive design features, and that really does tie to how well the plant's designed and how much you can bound the maximum hypothetical or maximum credible accident. And it's just not clear to me how you can do that. That's where the power level is a little more attractive because that's a more direct bound on things like radiological consequence.

MR. BALAZIK: Yeah, this is Mike Balazik, NRC staff.

One thing I want to mention is one of the aspects of the rule is you have to come in with a final design upfront. Even though we do issue a CP and an OL separately, you come initially in with the final design. So, you should have all that information at that time, is how we structured the rule.

MR. DICKSON: We fully expect that the rule is going to drive design changes for some of these reactors in order to utilize the framework. It is difficult putting together acceptance criteria -- I mean, given the broad use cases, reactor types, fuel types, coolant types, the individual risk profiles. So, this is our initial attempt of presenting what we think can work.

And then, also, working within, again, that very important SRM-SECY-98-144 for performance-based and risk-informed regulations and guidance, we're trying to move away from very prescriptive requirements, and hence, why you do not see a power level in this rule.

MEMBER KIRCHNER: But 10 megawatts is prescriptive -- 10, sorry, metric tons.

MR. DICKSON: Right. And I'll talk a bit about that, how we came up to that as, effectively, a screening criteria for reactor designs that we've seen before. I studied that GE PRISM Staff Safety Evaluation, a modular high-temperature gas-cooled reactor, the recent work with NuScale. All these reactors have these performance-based features of intrinsic safety and long-term passive cooling.

With that, we went and took a look at how big these cores are, right? And for these reactors that have those types of design features, that type of performance, the numerical value of that core shook out to right around 10 metric tons uranium.

And I'll talk a little bit about that analysis, of how we got there, but this is the staff trying to come up with these more performance-based, less-prescriptive requirements.

DR. BIER: This is Vicki Bier.

MR. DICKSON: Uh-hum.

DR. BIER: I don't get us too far off-track at this point. So, you don't necessarily need to answer this right now, but let's put it on the table.

One of the things that I'm concerned about is the interpretation of "similar risk profile."

MR. DICKSON: Right.

DR. BIER: And if, you know, different microreactors may have new different risk levels, depending on details of the design, and do they have to be similar to a microreactor? So, I'll just table that for now.

MR. DICKSON: I think we could talk more about that as we go through the slides. But I'm definitely anxious to hear what the Committee --

DR. BIER: Okay.

MR. DICKSON: -- has to say and their deliberations, and all of this.

MR. BALAZIK: And Mike Balazik again, NRC staff.

One thing I would like to add is, early on, we discussed power level being an entry criteria, but we were thinking that it would be arbitrary. You know, we would pick it and we don't have much of a basis for it.

And, I mean, recently, the staff issued the NPUF rule, which we had designated a research reactor and testing facility, but two different power levels. We recently changed that to get rid of the power level and say, hey, you transitioned from a research reactor to a testing facility and that has

different licensing requirements. So, we made it more performance-based when we used the 1 rem criteria.

So again, that's why we didn't really want to use a power level, and Elijah will talk about how we ended up at the 10 metric tons of heavy metal.

And -- no, that's it.

(Laughter.)

CHAIR HALNON: Yeah, this is Greg.

Let's get into the details of each. We can probably circle back and talk more about some details of those.

MR. DICKSON: Yeah. I suspected that this part of the presentation would take the most time.

So, let's go on to slide 7, please.

All right. So, as I mentioned before, it is very difficult to devise a one-size-fits-all solution for the use of this rule. And this is the staff's presentation, what we think would be appropriate, based off of our understandings of the AEA, the Executive Orders, the Commission policies that are out there.

And so, first, what we're going to do here is talk very briefly about each of these, but then we're going to go into detail in each of them.

So, first, we have the accident

consequence criterion of 57.25(a) that requires a nuclear plant to demonstrate less than 1 rem TEDE, or Total Effective Dose Equivalent, to a member of the public in the unrestricted area for the duration of a broad range of design basis accidents.

Second, we have the fuel mass limit of 57.25(b). This caps the inventory of heavy metals. Specifically, it's the fuel in the reactor core producing power, not fuel that might be in a reactor core as storage. So, there are some designs that have an in-core storage. So, this would be limiting the potential source term and bounding accident consequences without requiring additional performance-based requirements or severe accident analyses.

Then, we have the third. The rule requires the reactor to satisfy six design criteria attributes under 57.30, covering reactivity control, heat removal, fission product retention, shielding, effluent control, and security by design. These attributes reflect features commonly associated with inherent and passively safe microreactors and other reactors of low consequences.

So, the takeaway is that these eligibility elements together serve as the gate for making

high-volume licensing feasible under the Part 57 framework, and in doing so, we're supporting the Commission's broader expectation of advanced reactors. You can read about this in the Advanced Reactor Policy Statement of passive safety, higher reliability, design simplicity, long coping times, and reduced accidents through prevention, as well as predictable behaviors.

So, before I go on to the following slide, there's also another really important aspect I would like to bring to the Committee's attention for Subpart B. It's that Subpart B does play a crucial role in strengthening the public's communication -- it does play a crucial role in strengthening the public communication by presenting advanced reactor designs in a way that promotes broad understanding and to support informed acceptance.

As these technologies become more common, it becomes vitally important to provide simplified explanations to anyone, regardless of background or education level, so that they can clearly grasp what the design is and why that matters.

And we'll talk at a very high level about each of these design criteria attributes in that context as well.

So, the main point of the idea of the entry of the eligibility criterion is that a well-designed advanced reactor should be so safe and robust that, even under accident conditions, radiation exposure to the public would be limited to 1 rem, which effectively doesn't have any measurable health effects.

CHAIR HALNON: Elijah, I mean, on the previous slide, you said you weren't defining "microreactor" and "low consequence." Yeah, you actually are by this criteria right here.

MR. DICKSON: The design criteria attributes?

CHAIR HALNON: Have you thought about just taking the word "microreactor" out of that stupid definition and just --

MR. DICKSON: Yeah. People focus on, or folks focus a lot of time and attention on the first two entry gates, right, the 1 rem and the material mass limit, it's the design criteria attributes that are going to truly inform design changes.

CHAIR HALNON: I guess my point is that we've got large light water reactors; we've got SMRs, and now, we're defining something less than that, or at least it feels that way.

MR. DICKSON: Yeah.

CHAIR HALNON: You have the test reactors and other things. If you don't attempt that and just take the word "microreactor" out of the title, and leave it at that, now people won't fixate on another class of reactors, but now you just have reactors that are --

MR. DICKSON: Of low consequence.

CHAIR HALNON: I mean, just the thought process, I think one of the Commissioners came up with that.

MR. DICKSON: Yeah.

CHAIR HALNON: It makes a lot of sense.

MR. DICKSON: Low consequence by design.

CHAIR HALNON: Something to that effect.

MEMBER KIRCHNER: Elijah, you made a statement with regard to the fuel mass limit. Do you want to repeat that again?

MR. DICKSON: Sure. So, the fuel mass limit, it caps inventory at 10 metric tons heavy metal, limiting the potential source term, the decay heat profile, and bounds accident consequences. And we think that this does reduce the requirements for severe accident analyses.

MEMBER KIRCHNER: That's not a given,

though, with that.

MR. DICKSON: With the design --

MEMBER KIRCHNER: In fact, the concern that we should have is the proliferation of risk. Ten metric tons, hundreds of megawatts, has the potential for severe accident consequences. And just because it's 10 tons doesn't mean there isn't the potential for a severe accident.

MR. DICKSON: Yeah, we have a question in the FRN talking to this about the need of --

MEMBER KIRCHNER: Because the site could have 10 of these reactors --

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: -- and you have a cumulative, you know, a cumulative source that's quite significant if you had a common-cause event.

MR. DICKSON: We have in the guidance document discussions in regards to how you come up with your MHA or MCA --

MEMBER KIRCHNER: Right.

MR. DICKSON: -- and that an important aspect of how we design the first entry criteria is that it is on the plant basis, not on an individual reactor basis.

MEMBER KIRCHNER: Right.

MR. DICKSON: And so, we would be searching for common causes between reactor designs if they do have safety-related SSCs that are shared, but that would be defined in scaling that site boundary.

MEMBER KIRCHNER: But again, an unintended consequence may be the following: applicant A comes in with the first-of-a-kind --

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: -- or nth-of-a-kind at one site.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: Decides, oh, I need nine more of these, or five, or six. In fact, we've actually heard about strategies --

MR. DICKSON: Yeah.

MEMBER KIRCHNER: -- that are like that.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: So, you're not really capping the total source term with 10 metric tons per se, if there are multiple units at the same site.

MR. DICKSON: That's correct.

MEMBER KIRCHNER: And you're not, by definition, on severe accidents. If you argue with your attribute list that I'm reducing the potential for a severe accident, you know, that's a good,

logical approach.

MR. DICKSON: Okay. I appreciate that.

MEMBER KIRCHNER: But heavy metal loading is not going to eliminate the potential for severe accidents.

MR. DICKSON: Okay.

MEMBER KIRCHNER: And we will be concerned, even to 1 metric ton.

MR. DICKSON: Uh-hum. Okay. Noted.

Okay. So, on to slide 8. So, this slide covers the first of the two eligibility gates for entering Part 57.

The dose-based entry criterion under 57.25(a), this is the foundational screening that ensures that only reactors with inherently low offsite consequences can use the streamlined Part 57 framework.

Applicants must demonstrate that, for a postulated accident, which bounds a broad range of design basis accidents, that no member of the public will receive a dose of greater than 1 rem at that site boundary. We do not define what that accident is in the regulation, but in the regulatory guidance -- we will talk about that soon -- that we want to utilize either maximum hypothetical accident methodologies or

maximum credible methodologies.

And there is a process in which you would assess credible events at the facility, understand how your facility responds to these different events. You then take those events as design basis events and you put them into like categories as design basis accidents. Right?

And then, from there, you would identify either your maximum credible accident -- maybe take another failure or assume something else happens at the plant -- or you do the maximum hypothetical accident approach at that point.

So, going through that logical process helps you decrease the uncertainty in the source term that you would derive for siting your site boundary.

Again, that's all executed in the guidance itself.

VICE CHAIR PETTI: So, Elijah, are we going to talk more about that in subsequent slides?

MR. DICKSON: Yeah, we will.

VICE CHAIR PETTI: And then, the guidance?

MR. DICKSON: Right, right.

VICE CHAIR PETTI: This is the case where it was at least confusing to a couple of us. Because I know some of this stuff looks awfully familiar to

NPUFs --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- and research reactors and that guidance.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: The words may mean different things to different people when you read them.

MR. DICKSON: Yeah. Okay. All right.

So, from there, we can go ahead and transition on to the next slide.

I would also like to mention that we have used a dose acceptance criteria for a very long time, 70 years now at this point. So, it's very similar to what you find in 10 CFR 50.34 and the 25-rem dose acceptance criteria, or the 1 rem that's there for the NPUFs.

MEMBER KIRCHNER: I think here, in effect, you're reverting to 10 CFR 50 as a basis, as you think about it. Because 10 CFR 50 requires you to, essentially, postulate a maximum hypothetical accident. And that, at least for the LWR technology, has been defined, you know, through the old TID and other updates.

MR. DICKSON: Yeah, that's right.

MEMBER KIRCHNER: And that is a severe accident.

MR. DICKSON: That is a severe accident.

MEMBER KIRCHNER: And then, you're not doing a mechanistic source term anymore. You're just doing a mechanistic dose consequence analysis.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: Which I think is -- just one member's opinion -- appropriate here if you're trying to bound something for a large class of reactors --

MR. DICKSON: That's right.

MEMBER KIRCHNER: -- so you don't know what the technology is.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: And you don't -- I don't think you want to be in the situation where it's a question of sharpening the pencil on some mechanistic source term calculation.

VICE CHAIR PETTI: I would just note, the 1 rem is also EPA protective action.

MEMBER KIRCHNER: Right.

VICE CHAIR PETTI: So, the 1 rem is also the EPA protective action guideline, which is something, at least in the advanced reactor community,

has always been a goalpost. So, there's some alignment here.

MEMBER ROBERTS: This is Tom.

I would also caution that the maximum credible accident is not the same as a maximum hypothetical accident. And I want to compliment staff for doing a great job in the draft guidance of filling out the history on both terms and the somewhat tortuous history of them.

But I don't think you'll wind up in the same place necessarily as 10 CFR 50 when you do a maximum credible accident per the current guidance. And that's coming up in a slide, but I just wanted to caution.

I'm not sure I agree with the analogy, at least as I understand it, for 10 CFR 50, but I think it could be closer to a mechanistic source term, where you use the maximum credible accident approach from the direct guidance.

MEMBER ROBERTS: It could be.

VICE CHAIR PETTI: Yeah, also, I can remember sitting here asking other staff about what's the difference between an MHA and an MCA, and their not wanting to touch that at all. I give you guys a lot of credit for really trying to articulate --

MEMBER ROBERTS: Right.

VICE CHAIR PETTI: -- some clear differences. I thought it was well --

MEMBER ROBERTS: Thanks.

MR. DICKSON: We have a slide on that, and I'll talk a little bit more about those concepts and why one designer may choose to do one over the other, and how that could help their business case or not help their business case. It's a pretty important initial decision in regards to siting these facilities.

MEMBER KIRCHNER: Well, this is another area, Elijah, where I'm sure you're anticipating there would be quite --

MR. DICKSON: Yeah.

MEMBER KIRCHNER: As Tom points out, the MCA might be closer to a mechanistic source term approach, but, okay.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: I'll let you go on.

CHAIR HALNON: Go on.

MR. DICKSON: Okay. So, we'll talk about the fuel mass limit here on this slide.

So, this slide covers a second eligibility gate of the Part 57 framework, the fuel mass limit.

And I'll talk a bit more in-depth in the discussions that we had in The Federal Register of the staff's analysis of coming up with this numerical value.

So, why did the NRC select the material mass limit? We did not want to include an arbitrary power level, since it is clear that some reactor designs can achieve higher power levels safely than other designs.

Some designers do have business cases which utilize highly robust, but very expensive fuel, but they do exhibit pretty exemplary safety profiles; some of these reactor cores do.

So, the fuel mass limit is based on experience of past reactor reviews. I had mentioned, like, the modular high-temperature gas-cooled reactor review, the GE PRISM review, as well as the large swath of LWR analyses that we have done.

And it, effectively, serves as a deterministic screening criteria for the rule. It's not a performance-based one; it's a deterministic screening criteria.

So, we would like to point your attention in The Federal Register Notice to table 1, which presents our assessment of several reactor designs, ranging in power level from just a few watts up to

several thousand megawatts.

The evaluation included several non-LWR reactor types with various fuel forms, including TRISO, oxide, molten salts, as well as coolants -- gas, molten salt, liquid metal, as well as water reactors.

And from there, we also looked at these characteristics of their safety profile, the things that they rely on to handle that decay heat, when they do shut down, and can they handle that decay heat when it's shut down passively or not? Do they need to rely on these systems?

So, to understand the various reactor technology safety profiles, we did review several published scientific studies. We looked at staff's preliminary Safety Evaluation Reports and lots of environmental review reports.

The review focused on identifying common design attributes among these reactors, such as strong negative reactivity feedback, robust fuel forms, higher thermal margins, and passive heat removal that inherently limits transients and accident progression.

The NRC found non-LWR designs in microreactors are often designed with large thermal capacities, which allow them to dissipate operational

and decay heat passively for relatively long periods of time without the need of active systems or operator actions.

The designs also feature large shutdown reactivity markings and other intrinsic safety characteristics that provide strong inherent barriers to accident progression.

The overall result of these safety behaviors can be well-understood without relying on sophisticated probabilistic analysis methods. You can look at engineering design. You can see the margin and do confirmatory analyses and make your safety decisions based off of, effectively, engineering design.

So, based on the evaluation of special nuclear material inventories in this analysis, and these characteristics of non-LWRs, small LWRs, and the very large LWRs, we have concluded that establishing a defined special nuclear material limit would be technically justified as an entry criterion for Part 57.

The material limit would be defined as the total inventory of thorium, uranium, and plutonium contained in the reactor core not to exceed 10 metric tons.

The evaluation showed that the designs within this material limit likely have inherent passive safety features and exhibit favorable safety profiles, despite the variations in core design and thermal power levels.

So together, these insights support the NRC's determination that a numerical material limit, based off of this assessment, due to the inherent and passive design features of these reactors that we looked at, can be part, in effect, of a regulatory threshold or a licensing approach, and can help enable the rapid efficient licensing of microreactors and other reactors with comparable risk profiles.

Often, you know, we do think about, and we did focus on, reactor designs that have high enrichments. They have high power densities, high linear heat generation rates, but there are other classes of reactors that don't have those types of characteristics for compact core designs that weren't necessarily scoped into this analysis. You can think of designs such as the CANDU-type reactors, very large reactor cores, lots of special nuclear material in it, but either low or natural uranium enrichment -- natural uranium enrichment or low enrichment, and they do exhibit these low power densities, these low decay

heat profiles. Again, that's another thing to think about as we move through this process.

So, with that, we understand that this fuel mass limit entry criterion has a lot of questions and there's a lot of concerns. And so, we have proposed a question in The Federal Register as question No. 1, as to whether an alternative performance-based acceptance criterion would be appropriate.

Examples such as an adiabatic heat-up rate calculation, we do those for spent fuel pool cooling purposes for decommissioned facilities or a severe accident analysis requirement. To what degree that severe accident analysis is intended to be suitable for these very small reactors could make things a bit more complicated. Or should it be a combination of the two, some type of material limit and/or -- right, if you look at the rule? -- or a performance-based assessment that shows that your reactor meets these design criteria attributes of intrinsic safety and passive safety features?

All right. Any other questions on this? I think we've covered quite a few questions on this.

CHAIR HALNON: This is Greg.

I just wanted to ask the members. I think

we've fairly expressed the hesitation and concerns with the mass limit, but that question is for us at the end as well. So, let's be thinking about the other side of the equation and how can we suggest something that may be a pathway criteria.

MR. DICKSON: Uh-hum.

CHAIR HALNON: So, be thinking about some ideas there.

I like the idea of simplicity. The adiabatic heat-up rate threshold doesn't sound (audio malfunction) to me.

VICE CHAIR PETTI: It's certainly easier than a severe accident.

CHAIR HALNON: That's right. I'm thinking more like sizing --

VICE CHAIR PETTI: I think I can still do that on the back of an envelope.

CHAIR HALNON: -- this outside boundary where you get that rem, you know, 1 rem. Make it a quarter mile or something. You know, that's simple.

So, let's think through that for the rest of the day, and when we get to the deliberation, we can talk more about it.

VICE CHAIR PETTI: Elijah, just a question. You know, when we looked at it, we saw it

kind of as a deterministic kind of criteria --

MR. DICKSON: That's right.

VICE CHAIR PETTI: -- to balance against. The 1 rem, there's sort of a missed sense to that, right?

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: And so, the complementarity there, I like having -- we liked having one in each bucket, if you will --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- sort of as a backstop.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: But if it's the right one, though. Was that, guys, in your mind, as you were thinking about it?

MR. DICKSON: That's right.

VICE CHAIR PETTI: Okay.

MR. DICKSON: Exactly. That's exactly right.

We would appreciate it -- this would be a request from the staff -- that if you do recommend something, that there is a bases that we can go study and assess --

VICE CHAIR PETTI: Well, the letter says

power level --

MR. DICKSON: That's right.

VICE CHAIR PETTI: -- and the reason is because does, the isotopes that are most concerning in dose --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- are directly related to power, not to burnup.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: I mean, iodine-131 --

MR. DICKSON: Right.

VICE CHAIR PETTI: -- is the classic. If any of the shorter-lived ones come out, they all saturate the power.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: You know, cesium and others are usually not as risk-dominate, particularly if they're TRISO-based. And a lot of the micros are TRISO-based.

MR. DICKSON: Right. That's right. And so, the release of iodine or cesium, it doesn't scale well between reactor designs themselves. You know, we know, for LWRs, when they melt, the iodine and cesium shoot right out of the fuel matrix, and then, become volatile, and that's something that we need to

address.

For the high-temperature, gas-cooled reactors, that matrix is pretty tight, and it's very hard to melt that fuel.

VICE CHAIR PETTI: And then, in the sodium system, the sodium will hold the cesium.

MR. DICKSON: That's right. Sodium --

VICE CHAIR PETTI: They're both liquid metals; they like each other.

MR. DICKSON: Sodium is, effectively, a getter --

VICE CHAIR PETTI: Right.

MR. DICKSON: -- for these fission products and it holds them.

MEMBER KIRCHNER: I would caution, without severe accident analysis, a good designer -- and certainly, a good regulator -- will look at the "what ifs?" With some of the SMRs, we've had extensive debates about cliff-edge effects, et cetera, to kind of put a (audio malfunction) on that performance/risk-based set of analyses that are done.

But I wouldn't lead to the implication that you don't have to do severe accident analysis. You almost always look at the "what ifs?" whether it's an MHA or an MCA, or a completely mechanistic

evaluation.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: We should avoid hubris at all in this going forward. Because it's small doesn't mean it's safe. That's a misnomer.

MR. DICKSON: Right.

MEMBER KIRCHNER: We've had small reactors that had severe accidents.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: SL-1, et cetera. So, you can't just a priori come to that conclusion. It's the totality of what you're proposing --

MR. DICKSON: That's right.

MEMBER KIRCHNER: -- with the attributes.

MR. DICKSON: With the attributes.

MEMBER KIRCHNER: Because the first two criteria, we can deal with the first one by having a large site.

MR. DICKSON: That's right. A very large site.

MEMBER KIRCHNER: Right. And so, it's just a caution.

MR. DICKSON: Yeah. Understood.

MEMBER KIRCHNER: Yeah.

VICE CHAIR PETTI: We do think those

attributes are the right ones --

MEMBER KIRCHNER: Yeah.

VICE CHAIR PETTI: -- from my --

MR. DICKSON: Okay. Thanks.

VICE CHAIR PETTI: You know, I think that's --

MR. DICKSON: Yeah. Okay.

So, slide 10.

Slides 10 and 11 introduce the six design criteria attributes the applicants must demonstrate under 57.30. These criteria form the technical backbone of the Part 57 framework. Together, they ensure that any reactor license under Part 57 has core safety characteristics consistent with inherently low consequence designs.

Also, as I had mentioned before, they play an important role in risk communication to the public concerning our regulatory decisions. It is important to facilitate rapid deployment, and therefore, it's imperative that the public do not feel threatened by these types of technologies. So again, it's a very important communication tool.

So, we'll talk about each of them. The first three are here: 57.60(a), (b), and (c).

So, the first one is reactivity control.

This criterion ensures that the reactor can control power during normal operations, rapidly insert reactivity control devices to achieve and maintain a safe shutdown state under accident conditions, and exhibit net negative reactivity feedback with increased power. These features reflect the expectation for inherently stable, self-limiting reactor behavior.

(b) Heat removal. The design must prove highly reliable passive decay heat removal during accident conditions. The requirement emphasizes past capabilities to hold fuel and coolant temperatures within design limits to protect the fuel, the coolant boundary, and fission product boundaries. This reinforces the broader Part 57 philosophy of reducing reliance on operator actions and active safety systems.

We also then have fission product retention. This attribute focuses on maintaining the integrity of the engineered fission-product barriers during both normal and accident conditions. The intention is to limit radioactive releases by design rather than through mitigation or procedural.

Any questions so far?

MEMBER KIRCHNER: Elijah, do you have --

this is Walt Kirchner.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: Do you have a definition for safe shutdown that you're using? Because this is one that we've seen some bad interpretations.

MR. DICKSON: Yeah. I believe --

MEMBER KIRCHNER: Does this include adiabatic heat-up rate --

MR. DICKSON: The definition is, in part -- yes. The definition is part of the requirements, 57.3.

I need to go back and look myself, too. But I'm almost certain it's there.

MEMBER KIRCHNER: Well, we've had some discussion with applicants, without going into detail, that suggest, as long as fuel and temperature conditions are within design limits, that they don't necessarily have to be subcritical. So, that's why I'm testing.

MR. DICKSON: Is that so the -- is that like -- I'm trying to think, in practicality, what that means. Is it so they can write --

MEMBER KIRCHNER: Keeping the rods in or control drums --

MR. DICKSON: That's right. So, they can

write out transients, is what --

MEMBER KIRCHNER: They're suggesting that can write out transients --

MR. DICKSON: That's right.

MEMBER KIRCHNER: -- without being subcritical.

MR. DICKSON: Uh-hum. Understood.

MR. BALAZIK: This is Mike Balazik of the staff.

Yes, we do have a safe shutdown, and it's defined as, "under design basis accident conditions with the loss of emergency power and offsite power, bringing the nuclear reactor to safe, stable conditions specified in the plant technical specification." So, we have a definition there.

MEMBER KIRCHNER: Yeah, but does that mean and include subcritical?

CHAIR HALNON: No, it kicks the can down the road to the tech specs.

MR. BALAZIK: It does.

CHAIR HALNON: And that's maybe appropriate from the standpoint of whatever design you're dealing with, but I know --

MEMBER KIRCHNER: Hold on a minute, though.

So, you started out by wanting to have transparency with the public. Do you think a reactor is in a safe shutdown condition and it's not subcritical, and you don't have any control of criticality?

CHAIR HALNON: Well, I think that's a couple of questions. So, the last one is, you don't have control of criticality is a loaded question.

MEMBER KIRCHNER: Well, are you shut down or not? Is K-effective less than 1? It's a simple --

CHAIR HALNON: Well, that's a better question, is K-effective less than 1?

MEMBER KIRCHNER: So, is K-effective less than 1?

MR. DICKSON: Yeah, that's the intent and that's what you find -- that's what you find in the guidance document.

MEMBER KIRCHNER: I'm just checking. All right.

MR. DICKSON: The reactivity control section under the guidance document, that you set down --

MR. DICKSON: Okay. Well, I mean, that's something that we can take back and look. If the requirement needs to say something like that, we can

definitely assess.

VICE CHAIR PETTI: Well, there's a SECY. Is it 94 or 93?

MEMBER KIRCHNER: It's 94 -- 93-084 or 94-087, or something. I'm sorry, I didn't look it up in advance.

VICE CHAIR PETTI: Because of the advanced reactors have some of the odd characteristics, that it's sometimes hard to get subcritical.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: Well, it specifically addressed MHTGRs and sodium-cooled reactors. Because, understandably, you don't want to cool down the sodium reactor --

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: -- and have freezing in the system.

MR. DICKSON: Yeah, really. That's correct.

MEMBER KIRCHNER: That introduces another set of potential accident scenarios or problems with control surfaces, or whatever. So, that SECY did provide for them being at kind of a hot standby condition.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: But it, to my recollection, definitely said that you were subcritical.

MR. DICKSON: The intent was not to be at hot standby. It was supposed to be subcritical.

VICE CHAIR PETTI: So, if you go to the Reg Guide 1.232, MHTGR Criteria 26 and 27 --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- and read the rationale, that where the SECY is. So, you can find that easily.

MR. DICKSON: All right.

VICE CHAIR PETTI: I don't remember the number. Because the rationale is in there. Things are in the SECY, and that's how you can get to it easily. I was looking at it yesterday.

MEMBER KIRCHNER: Well, I'm thinking, you know, again, for the public perception and confidence, we want to know that the reactor is shut down; that is, it's subcritical, especially if autonomous or remote operations are being considered.

MR. DICKSON: Understood. Okay.

MEMBER KIRCHNER: So, I think attention in this area is potentially needed.

MR. BALAZIK: Yeah, this is Mike Balazik

with staff.

I will say that we're seeing a lot of designs out there that actually are driven by accelerator. So, they actually operate in the subcritical region, and that these may fit in this rule, those types of designs.

VICE CHAIR PETTI: Accelerator-driven systems have a whole set of issues, and DOE has looked at these things a lot.

MEMBER KIRCHNER: And the military.

VICE CHAIR PETTI: You just have to be careful; the advocates of accelerator-driven systems tell you that, you know, they're the absolute perfect solution, but the safety issues are very similar, whether you sit on one side of the K-effective equal to 1 or the other. You end up bumping into all the same issues.

MR. BALAZIK: No, agree, and that's how we -- I know there are some members here -- that's how we license SHINE. SHINE, I mean, it's a medical isotope commercial facility, but, you know, it operates at below K-effective of 1. And we had that framework in place for those types of facilities -- somewhat.

MR. DICKSON: All right. So, we can go on to slide 11.

All right. So, we'll talk about the other three design criteria attributes.

We have shielding. The shielding criterion requires adequate permanent and temporary shielding to meet the Part 20 limits for workers and public protection. This includes operation, shutdown, transportation, and under abnormal conditions, meaning shielding must be robust and maintain its integrity during these off-normal events.

We have reactivity control. The reactor must be designed to meet the Part 20 normal operation effluent limits. This ensures that the effluent controls are built directly into the design rather than left purely to operation and programmatic programs.

And then, we have, lastly, security by design. Finally, the design must integrate security by design, resolve insecurity concerns through engineered features, where possible. This aligns the safety and security requirements and reduces the need for compensatory administrative or operational measures.

So, in closing, these six design criteria attributes, when paired with the eligibility criteria themselves -- again, looking at them as a whole --

collectively, they define the type of reactor that could use the Part 57 streamlined risk-informed and performance-based framework. This ensures the safety is fundamentally embedded in the design rather than depending on the complex system and operator actions.

And with that, I can take questions on slide 11.

VICE CHAIR PETTI: Yeah. Elijah, just for clarity, you know, in the small micros, they're neutronically incredibly leaky.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: So, neutrons are going everywhere, and you're going to activate a lot of air. Is that what you mean by "the radioactive sources"?

MR. DICKSON: Yeah, that's right.

VICE CHAIR PETTI: The argon --

MR. DICKSON: That's assessed. The argon is assessed in Chapter -- I think it's 11 is the radiation protection and radwaste, that you need to assess the argon and the other --

VICE CHAIR PETTI: The other issue is activation of the dirt.

MR. DICKSON: Yeah. That's right.

VICE CHAIR PETTI: And the mediation. I mean, leaving that footprint, if you will, if they're

mobile, you know, take them away.

MR. DICKSON: Uh-hum. And tritium, of course.

VICE CHAIR PETTI: And tritium, even though it's probably not -- I mean, you know, the public safety regs are fairly high --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- but the public gets very alarmed at --

MR. DICKSON: That's right.

VICE CHAIR PETTI: So again, the shielding, that's why I was really happy to see the shielding. It's a constraint that if you haven't been in the reactor world and you're doing the big ones, and you go to the small ones, it's a different constraint.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: And it really does change --

MR. DICKSON: Absolutely.

VICE CHAIR PETTI: -- how you have to think about design.

MR. DICKSON: Yeah, about the bioshielding, about all of that.

VICE CHAIR PETTI: Right, right.

MR. DICKSON: Exactly right.

VICE CHAIR PETTI: Particularly, if you wanted to do mobile, you know.

MR. DICKSON: Right.

VICE CHAIR PETTI: So, let me ask about security by design, without getting into a lot of details, because it's not our swim lane.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: What I worry about is there are bunch of design basis threats over here that we don't talk about. We've done a really good job at trying to put a cap within MHA on the traditional safety stuff. But what if the security requirements kind of trump all that and require some scenario that, you know, the designer is going to go, "We don't think that's credible." And they're going to say, "Well, no." You know.

I mean, the alignment of DBAs and DBTs, if they get out of alignment --

MR. DICKSON: Yeah.

VICE CHAIR PETTI: -- something is going to drive the facility design.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: And again, it's not our swim lane. I just worry in the back of my mind that

we've done so much good work here on the licensing basis and the DBA, making sure that that stays consistent when you integrate security.

MR. DICKSON: Understood. There is the design basis threat analysis that comes with this framework. That's about all I could really see on that.

CHAIR HALNON: Yeah, Elijah, this is Greg.

When I look at the criteria, the first three, (a), (b), and (c), have to do with what you can put on a trailer and transport.

MR. DICKSON: Uh-hum.

CHAIR HALNON: The last three have to do a lot of the site itself and where it's going to end up.

MR. DICKSON: Right.

CHAIR HALNON: How do you reconcile the thought process that these things are going to be movable to different sites? And how do you license a site that you don't even know where it's going to go? Or possibly, a site that you continue to load more on there.

So, we have one design that we're kind of cogitating on is that they may have seven cores, six of them spent and one of them still operating on a

site. So, how do you reconcile all that in a licensing process and saying it's a final design?

MR. KENNEDY: This is Duke Kennedy of the NRC staff.

So, I'll talk a little bit more about this later when we talk about multi-plant sites and the different licensing pathways.

But, essentially, the idea is that, for sites where you might have reactors that are operated, then taken out of service, replaced with new ones, the old ones might stay at the site for a while to cool down. The idea is that all needs to be considered upfront and a plan incorporated into the Construction Permit application for how many reactors can be at a site in various states of operation or shutdown.

And so, it's really, when you get into those licensing frameworks with more flexibility, we just don't have surprises as we go on. So, we want the developers to do the best they can to lay out their entire deployment process upfront, so that can be reviewed.

It can always be amended, you know, to match real circumstances, but the idea is to get as much of that upfront information as possible, so we can make a decision.

CHAIR HALNON: Okay. Well, when you license these, go on through this criteria, you're looking at potential sites that it could go. I mean, you have to license the site as well, and these are transportable.

So, I mean, do you anticipate someone coming and saying, "We may be transporting it to X and now we're going to go to Y. And we may not even be at Z for very long because the use may not be necessary."? I mean, all that has got to be considered upfront, and all those sites have to be looked at. And that's what is going to run through this criteria then?

MR. KENNEDY: Yeah, that's correct. So, each reactor has to be covered by an Operating License, and that Operating License is site-specific.

CHAIR HALNON: And they're connected to a site then? It's not necessarily just an Operating License for a reactor plant?

MR. KENNEDY: That's correct. So, the Construction Permit, there's an option under a Construction Permit to specify a large area within which the reactor could be deployed.

But before the NRC authorizes that reactor to be operated, it will need to issue an Operating

License for a specific location.

CHAIR HALNON: Okay.

DR. BIER: We can, again, discuss in more detail when we get to the multi-reactor sites.

This is Vicki Bier.

Maybe not immediately, but someday one could envision sites just like an industrial park that may have multiple reactors, each of which is owned by a different company. The user companies own their own reactor, but they're all collocated. And so, that, again, may create complications for the licensing, and if the licensing is based on the output rather than on the site, you're kind of encouraging that type of development, like, oh, I can get around this by licensing in different ownership. So, it may happen someday.

MEMBER KIRCHNER: Just on the last one, security by design, it's something that we have included. It was in the Advanced Reactor Policy Statement, of course, back in 2008. But have you thought any more about what that really means?

I mean, as Dave said, we have, the Committee has direction from -- I should start with this -- through a SECY. Our Charter doesn't include looking at security, but it does include looking at

consequences.

And so, have you thought any further about, what would you be looking for in security by design? Is this just a throwaway tagline or do you have something specific in mind?

I can give you some examples, but if you look at just fuel forms, start with fuel, a liquid fuel system, that's going to raise a lot more concerns -- let me put it that way -- than a solid fuel system. Ceramic fuels are first order less useful and easily diverted and reprocessed. Metallics are much more readily reprocessed, and so on.

So, do you have some hierarchy in mind? Do you have some kind of real design requirements that you're looking for?

MR. DICKSON: There's regulatory guidance on this, I believe. We do have RSL and security here, but I would think you would start --

MEMBER KIRCHNER: That's the applicants. You know that.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: So, from the public format, or forum, what kind of requirements can you put out there to guide them to say, "Yeah, I have included security by design."? What have I done?

MR. DICKSON: So, the requirements are under 57.60(a)(8) for operational programs. There is a security requirement there that, then, kicks you out to the other requirements forming those types of analyses.

MEMBER KIRCHNER: That's more the operational.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: I'm thinking more of reactor design.

MR. DICKSON: Yeah.

MEMBER KIRCHNER: Anything, in particular, that you can put out there in guidance, even if it's not a requirement? Because this sounds good, but what does it practically mean? And what would you, as the staff, be looking for in a design to say, "Yeah, check that box. Then you've addressed that."?

MR. DICKSON: I mean, of course, insider threats and things of that nature. You would also want to look to see how they perturb the thermal-hydraulics of the reactor for these passive features, right? Can you stop those thermal-hydraulics from doing the things they need to do to cool the core? Those are the types of things I think would be looked at.

I know Argonne National Lab is actually doing research in this area, security by design, looking at different ways that you can overcome the thermal-hydraulics of a nuclear system to keep them from performing their function. There is research in that area.

And I got --

MR. ULSES: Actually, Elijah -- to answer your question, I'm Anthony Ulses, the technical advisor for security.

The answer to your question is, yes, sir, we have thought about this. It rapidly gets into topics and scenarios that it's very difficult for us to discuss publicly, but we have given it some thought.

What this really is intended to kind of drive, as the members have discussed, this is really, ultimately, a design optimization process. We expect, for certain reactor designs, there may be some scenarios where the designer will not be able to rely on the inherent physics of the reactor, for example, to mitigate the consequences of a security event, which is why we have included in the rule framework the possibility of having the need for an operational security program. And that's a structure that we set

up very consistent to what you see in Part 53, for example, which the Commission has recently released.

So, to answer your question, yes, we have considered it. To the extent that we can provide guidance, we've tried. And as we work with applicants, the expectation is they would come into the Commission and we would establish the necessary programs to then have access to safeguards information, which will allow us to give them the information that they need to perform a fulsome design.

CHAIR HALNON: Okay. Let's leave that discussion here and table the rest of the security discussion. We're getting beyond our scope at this point.

I appreciate the questions, but let's move. Let's get through the construction next session, and then, we're going to take a break.

MR. DICKSON: So, move to slide 13.

At this point, we're going to talk about -- okay. All right. So, slide 12.

So, this slide explains the Part 57 structure for the traditional licensing process for a joint Construction Permit and Operating License. I think I, more or less, covered most of this, these

items, through the question-and-comment period at this point.

The technical information you can find in 57.60(a), which would be equivalent to what you find in Part 50.34.

So, the applicant must provide the Final Safety Analysis that covers the site characteristics, reactor design, accident evaluation, assessment of safety-related systems, the presentation of their PDCs, their Principal Design Criteria, as well as all of their operational programs -- from security, quality assurance, fire protection.

Reflecting that, the expectation is that they'll have operational readiness when they do submit an application into Part 57.

Yeah, please.

MR. KENNEDY: This is Duke Kennedy of the NRC staff.

I just want to highlight the top bullet here, "Front-loaded review," since the question came up about the level of information that would be provided.

So, this rule is for final reactor designs. This is not a rule that's for designers, that preliminary information, and get started doing

something while they finish their design. That's not what this is for. This is a rule to go quickly, based on higher level of certainty from the outside.

And so, the joint application shouldn't be confused with a joint Construction Permit and Operating License. It is two separate things: the Construction Permit, and then, an Operating License. The joint application just means that you need all of the information upfront that would, typically, be provided under Part 50, for example -- first, in the CP application, and then, the rest later in an OL application.

So, we use the joint application to just say everything for both of those, the license and permit, has to be provided upfront. And so, that, again, enables us to potentially have one contested hearing and potentially one ACRS review, as a way to save time and overall resources. For going quickly on designs where we have all the information upfront, we need to make all of our safety findings.

CHAIR HALNON: This is Greg.

Do you anticipate expanded limited work authorization requests, or have you looked at that regulation to permit? Because if the reactor plant is being advocated offsite, there's a lot of stuff onsite

that you can do to be ready for it. Is that part of the discussion?

MR. KENNEDY: There are no limited work authorizations under this rule. The idea is that the licensing timeframes are going to be short enough, onsite construction would be simple enough, that we wouldn't gain an overall decrease in the deployment timeframe by having this additional approval upfront. The same for early Site Permits.

CHAIR HALNON: Acting on simplicity and rapid deployment?

MR. KENNEDY: Yes.

MEMBER HARRINGTON: So, this is Craig.

Just to help me make a fine point on this, they would provide all the information, effectively, like today for an Operating License. They still might get a Construction Permit and an Operating License together or in some staggered basis, or something? But it would be one submittal, and then, it goes out. Is that correct?

MR. KENNEDY: Yes, that's correct. One application that allows us to, first, issue a Construction Permit after completing a full design review, a full review of operational programs, a full environmental review, and then, when it comes to the

point where they've actually constructed the reactor and have their operational programs in place, we have nearly an administrative function to actually issue the Operating License, because all of the review has been upfront, and it's just a verification that they're actually ready to operate that reactor. And that's what we need to issue our authorization, the Operating License.

There is an important feature of this rule for you to try to apply it to, like, a disaster relief scenario, where you need to get a reactor operating at a location very quickly. Because if you have a Construction Permit that already covers the area that that reactor might need to be deployed in, and we've reviewed everything, they can say, "Look, there's going to be a hurricane over here in a week. We potentially want to build a reactor here, construct it, which may just be dropping it off the truck." And then, we can very rapidly issue that Operating License in order to support those types of deployments.

MEMBER HARRINGTON: Okay. Thank you.

MEMBER PALMTAG: Yes, this is Scott. I just had a quick question on that.

This doesn't cover nth-of-a-kind? Every reactor is going to have to go through the same

process? If you have multiple reactors of the same design, how does that work? Each one has to get a CP and an OL?

MR. KENNEDY: No. Each reactor has to have its own OL, or Operating License, but one Construction Permit can cover any number of reactors that are specified in that Construction Permit. So, it could be --

MEMBER PALMTAG: So, one is going to be for design and an OL for each actual reactor?

MR. KENNEDY: Yes.

MEMBER PALMTAG: Okay. And this is just a minor clarification, but you're using the terminology "CP and OL," not SDA?

MR. KENNEDY: Yeah.

MEMBER PALMTAG: Not a SDA letter? No? Okay. But it's using the term for this "CP and OL"?

MR. KENNEDY: Yes. So, for actually deploying the reactor, we only have the Construction Permit and Operating License process. But related to getting some sort of design approval, we do have a Manufacturing License that will cover -- and we also have a Standard Design Approval, which allows for major portions of the design to be approved by the staff.

MEMBER PALMTAG: I guess I need some clarification. How does the SDA work? Explain how the SDA is different from the CP. Would it replace the CP for an nth-of-a-kind?

MR. KENNEDY: No. So, the Standard Design Approval would need to be referenced in a Construction Permit, in a joint application for a Construction Permit and an Operating License.

MEMBER PALMTAG: Okay.

MR. KENNEDY: So, if a designer has a Standard Design Approval for a major portion of their facility or for the whole thing, that can, then, be referenced by a joint applicant in order to get ahead and not having to necessarily re-review everything that was already approved in the SDA.

MEMBER PALMTAG: Okay. So, that's one way of getting to the nth-of-a-kind? You have an SDA, and then, the CPs. And that would also be fairly quickly for nth-of-a-kind?

MR. KENNEDY: Yeah. I'll go into more detail a little bit later, but yes.

MEMBER PALMTAG: Okay. Thank you.

MR. ARAGUAS: This is Christian Araguas.

I wanted to add -- and, Duke, I know you'll kind of probably cover this later -- and this

is in terms of the LWA. So, we don't offer the LWA, as you've heard, but we do offer a General License for construction activities. And we'll talk about the criteria for what would enable that for nth-of-a-kind.

MR. DICKSON: All right. So, we can be on slide 13.

We're wrapping up Subpart C, other things that would need to be incorporated into this application for a CPOL.

You need to have the environmental information that you can find under 57.60(b). The applicant would either justify the use of a categorical exclusion or provide an environmental report or environmental assessment. This creates a clear, streamlined environmental review path, depending on the deployment scenario.

There is also generic finality under 57.60(c). Further, an applicant can include a request for generic finality. If granted, issues resolved in the first application proceeding would carry forward to future proceedings.

This would allow later projects referencing an approved licensing basis to avoid going back and re-reviewing the design and all the operational issues. So, this would be narrowing the

scope of a review that it needs for an nth-of-a-kind type of review. And again, this is a key mechanism in nth-of-a-kind type of deployment models.

And then, lastly, we have 57.60(d), which is the large area designation. The rule allows for a joint CPOL application to designate large geographical areas where multiple reactors may be deployed. This can support flexible siting strategies -- for instance, for disaster relief and other time-critical applications.

And so, you just spoke to those two items in regards to what people are thinking in deploying these types of reactors.

MEMBER KIRCHNER: In a disaster relief kind of situation, one could see that the government would have a direct role, like FEMA, or whomever, DOE perhaps. In practice, wouldn't that -- if you have a hurricane in some location and it's declared a disaster area, whatever, wouldn't that be more a government activity, not private sector, although the private sector could be involved, obviously, in supplying the equipment, whatever?

MR. DICKSON: Sure.

MEMBER KIRCHNER: So, when you say, "large designated area," it seems to me this is harking back

to what we saw under 10 CFR 52, where, either through a design certification or an SDA, you've made a robust design where you know -- for example, on the seismic, it's a robust enough design that it fits within --

MR. DICKSON: That's correct.

MEMBER KIRCHNER: -- most geographical sites across the U.S.? That's what we've seen other vendors actually --

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: So, in practice, how broad a designated area? Are we talking west Texas to probably the Permian Basin?

MR. DICKSON: Yes.

MEMBER KIRCHNER: And so, how does the public get involved in that?

MR. KENNEDY: Yeah, this is Duke Kennedy of the NRC staff.

Can I delay that until later in the presentation?

MEMBER KIRCHNER: Yeah.

MR. KENNEDY: Because I will cover that.

MEMBER KIRCHNER: Thanks. Okay.

CHAIR HALNON: Okay. At this point, we're going to take a 10-minute break.

We've got quite a few slides left. We're

going to finish up at least for the morning session about 20 after 11:00. So, we'll have about an hour left to go. So, we'll need to be disciplined in our questioning, and whatnot.

I am going to say that the public comment period will start after lunch at some point, after we wrap up the rest of the slides, and whatnot.

And lunch will be complete at one o'clock.

So, that's kind of the lay of the land here.

So, it's 10:08 right now. We'll come back in session at -- yes?

PARTICIPANT: At 1:30.

CHAIR HALNON: At 1:30? Oh, I thought it was going to be an hour and a half. Okay. So, 1:30. We'll make that announcement prior to recessing for lunch.

But we're going to recess now for about 12 minutes and come back at 10:20.

(Whereupon, at 10:08 a.m., the foregoing matter went off the record and went back on the record at 10:20 a.m.)

CHAIR HALNON: Okay. We're back in session. It's 10:20.

And I'll turn this back over to Dave Petti

to lead us through the morning, the rest of the morning.

VICE CHAIR PETTI: Okay. It's going to turn to the Manufacturing License.

Are you going to go? Go ahead.

MR. KENNEDY: Okay. Thank you.

This is Duke Kennedy of the NRC staff.

Okay. So, the second licensing instrument that we have under Part 57, proposed Part 57, is a Manufacturing License.

This looks very much the same as a Manufacturing License under Part 53. So, it covers the final design of the manufactured reactor. It covers requirements related to quality assurance and some of the manufacturing processes, and it also includes provisions for loading fuel at a manufacturing facility, as long as the appropriate Part 70 license is also in play for the fuel.

A couple of things to highlight that are different than Part 53 or that there are provisions under proposed Part 57 where an applicant for a Manufacturing License can provide the final design information for the entire nuclear plant, and not just the manufactured reactor. And so, this is something a little bit different.

The staff can review the entire plant design and provide finality on that design. This is a way to accelerate nth-of-a-kind microreactor licensing, so that the entire approved plant design can be referenced in a joint application for a Construction Permit and Operating License. And so, it just allows a similar level of approval to what you might see in a Design Certification under Part 52 or Part 53, but through a different licensing vehicle, which is this Manufacturing License.

The technical contents of the Manufacturing License application are drawn from Subpart C on the contents for a joint application, but, of course, it doesn't include site-specific information.

And the Manufacturing License also allows an optional pathway for the applicant to get the operational programs for the reactor approved, to the extent that they've been developed without site-specific information at the Manufacturing License stage.

So really, through a Manufacturing License, an applicant can get their entire plant design approved, all of their operational programs approved, minus the site-specific information. So

that that's all been given finality and can be referenced in joint applications for deployment of the reactors.

Could we go to the next slide, please?

And then, of course, a topic of interest has been features to prevent criticality, which would need to be installed in the reactor before fuel would be loaded at a factory, and also, to be installed in the reactor before it can be transported loaded with fuel.

And so, there are provisions in the Manufacturing License, proposed Manufacturing License Subpart here, about those features to prevent criticality. It does not specify exactly what those features have to be. It could be a mixture of physical and administrative features. And so, that's something that's discussed a little more in the Implementation Guidance and something that we expect would be highly design-specific, and that we would have that information provided in the Manufacturing License application as part of the reactor design.

Next slide, please.

MR. BALAZIK: Good morning. Mike Balazik, NRC staff. I'm going to go over Subpart G.

Subpart G defines how Part 57 handles

irradiated fuel storage, decommissioning and license termination. The approach is intentionally flexible to accommodate a wide range of microreactor deployment models in business cases, including manufactured, transportable, and redeployable reactors that we've discussed at a high level earlier.

Part 57 relies on the existing 10 CFR Part 72 framework for the storage of spent fuel. Just like 50 and Part 52, Part 57 Operating License holders receive a General License for onsite storage of irradiated fuel. If the fuel is stored offsite, such as a manufacturing facility, then a specific Part 72 license is required for the storage of that material.

The rule also allows irradiated fuel to remain stored inside the reactor itself, provided that the conditions of the Operating License or any applicable Part 70 license continue to ensure safe storage of material or the fuel.

Decommissioning requirements under Part 57 are modeled after 10 CFR 50.82(b), which is for non-power reactors. What's unique about this, or I should say we included a lot of flexibility, that the applicant can submit the decommissioning plan upfront. So, they can have that assurance of decommissioning or they can do it later, when they cease plant

operations.

MEMBER KIRCHNER: Mike?

MR. BALAZIK: Yes?

MEMBER KIRCHNER: On that point, is there a potential problem? Say the applicant, the user, the utility, whoever it is, the private sector, goes bankrupt. We don't have a decommissioning plan. Do you have decommissioning funds?

MR. BALAZIK: Yeah, the rule requires decommissioning funds to be in place. That's within the rule, uh-hum.

MEMBER KIRCHNER: But if you don't have the plan, do you have enough funds in place to deal with the potential cleanup that's required.

Say the applicant deploys, contaminates the local ground and such. Since the NRC issued the license, who is liable in a case like that? Does it come back to the government?

MR. BALAZIK: Yeah --

MEMBER KIRCHNER: It's a policy issue maybe, not a technical issue, and may be outside our swim lane. But I think it's an important consideration, especially if you're dealing with contamination at the site.

MR. BALAZIK: Yeah, it's something we'll

look into.

MEMBER KIRCHNER: Thank you.

MR. BALAZIK: Yeah.

MEMBER PALMTAG: Since you brought up the decommissioning funds, how does the decommissioning funds work for a microreactor that could be onsite for a small amount of time? Do you have to pay all these funds upfront? Or over the perceived operational lifetime of the reactor?

MR. KENNEDY: This is Duke Kennedy of the NRC staff.

So, I think, in general, the decommissioning regulations under Part 57 provide all the same assurances that you would have under Part 50. There's some extra flexibility, I think, which is what we're explaining here, in order to provide more of that information upfront and to use more of the non-power reactor framework.

There is the ability that you have under Part 53 as well for site-specific decommissioning cost estimates, and those can consider things like, at the site, we're actually not going to dismantle the reactor; we're going to transport it away to a different license and we're going to dismantle it at a different facility under that other license.

So, you may have much less need for funds tied specifically to the operating site for decommissioning. But all of those funds need to be assured ahead of time, just like they do for current power reactors.

MEMBER PALMTAG: But you are considering a much shorter, a potentially much shorter lifetime for the reactors.

MR. KENNEDY: Potentially, yeah. Yeah.

MR. BALAZIK: Yeah, it depends on the business case and deployment model.

MEMBER PALMTAG: Thanks.

MR. BALAZIK: Yeah, I think we talked about that. Yeah, next slide, please.

CHAIR HALNON: All right.

MR. DICKSON: This is Elijah Dickson with the staff again.

So, Subpart I establishes how Part 57 addresses Transportation Package Design Certification, recognizing that many microreactor deployment models rely on transporting either the fueled reactor itself or the reactor as the content of a certified package.

Part 57 relies on existing regulatory frameworks under 50.71, which governs the packaging and transport of radioactive material. This ensures

continuity with established NRC transportation safety requirements.

There is risk-informed methodology options. Subpart I allows applicants to use a risk-informed method or other risk-informed approaches, provided the methodology has been endorsed and approved by the Commission.

The slide does cite a SECY paper on an example of such a risk-informed approach. And I think this was done for Project Pele.

So, for reactors as a package or the contents thereof, the rule explicitly supports two configurations: that the reactor fuel is transported as a package itself or is transported, again, as a certified package.

There are requirements for criticality prevention. So, regardless of the configurations Duke had mentioned, the transportation package or the reactor must incorporate features to prevent criticality during transportation. And this is essential to ensuring safe shipment of the fueled reactor.

So, in closing remarks on Subpart I, it recognizes transportation as a key enabler for manufactured redeployable microreactor concepts by

integrating risk-informed methods and flexible packing configurations. It supports a practical approach to licensing this part of the deployment cycle.

Any questions on slide 17?

(No response.)

If not, then we'll go to slide 18 and Mike has slide 18.

MR. BALAZIK: Yeah, I'm going to talk about -- this is Mike Balazik, NRC staff -- and I'm going to talk about Subpart B, which establishes the operator licensing framework for Part 57. It links operator requirements directly to the safety characteristics of the reactor design.

I do want to add that this is a little bit of a change, when we were with the Subcommittee, on how we approached this. So, some differences here I'll point out.

The key distinction is whether the facility is considered an operator-independent or operator-dependent facility. Those are two terms that we have defined in Part 57. And it's based on reliance on operator action to meet the 1 rem dose-based entry criterion of 57.25(a).

This is, I'll say, a concept that we brought over from Part 53. However, Part 53 uses a

little bit of a different terminology, where they use "self-reliant mitigation facility" and "interaction-dependence mitigation facility." And the reason why we didn't use the same terms is they have different requirements than we do. So, we had to come up with our own term, and that's where we came up with the operator-independent or operator-dependent facility.

All right. So, for operator-independent facility -- for reactors that do not rely on operator actions, the 1-rem criterion, Part 57 allows a General License for operators, a similar approach as Part 53. And these are designated, of course, as operator-independent facilities.

GLROs are authorized to manipulate reactor controls. Direct-licensed operators monitor they plant conditions during reactivity or power changes after completing facility-specific training and examination.

I'm going to go on to just human-system interface. The licensee must incorporate proper human factors requirements into the design, maintaining a staffing plan, and developing training, examination, and proficiency programs while using a systems approach to training. A lot of this is what we pulled

from Part 50.

Once individuals complete the initial training and exam program, the licensee may designate them as GLROs. The licensee must also report annually to the NRC with updates of the GLROs supporting inspection oversight.

So, the thought process here is that, once they meet the licensee's requirements, they can add in a piece of paper and say, hey, you're a GLRO for this facility. And they would submit that information to the NRC. So, when we go out and do oversight activities, we verify that they actually met those requirements.

On the next slide, I'm just go over specifically licensed operators. There it is.

Yes?

CHAIR HALNON: Before you go on, on that slide 18 -- this is Greg -- the very first bullet, no reliance to maintain below 1. So, if you have a reactor that doesn't get to 1 rem, but you have a lot of license basis accidents, you know, for the lack of saying, let's say .99 rem, does that mean that it can pass this criteria, even though the operators are having to respond to those license basis events?

MR. BALAZIK: Well, I'll say that, do they

have to take that action in order to prevent it from going over 1 rem? Is that what you --

CHAIR HALNON: I'm just saying that there's nothing it over 1.

MR. BALAZIK: Right. So, yeah, I mean, they would meet that criteria.

CHAIR HALNON: So, this is not even close to the self-mitigating facility --

MR. BALAZIK: Self-reliant?

CHAIR HALNON: -- of Part 53?

MR. BALAZIK: Right. Right. I mean correct. Then, that's why we came up with our own different terms than what was done in Part 53.

MEMBER KIRCHNER: But the problem here that I think Greg is leading you to is that the 1 rem has nothing to do with -- well, let me rephrase that. The 1 rem can be met by just being in the Permian Basin with a huge exclusion area or somewhere else in Alaska, or wherever.

MR. BALAZIK: Uh-hum.

MEMBER KIRCHNER: That doesn't tell you much about the reactor. It doesn't tell you much about controlling the reactor. It doesn't tell you much about the need for operators or not.

You're just saying, if you're below 1 rem,

you can use a GLRO. That doesn't make sense if you're talking about a large reactor of hundreds of megawatts. That just flies in the face of what's already out there in 10 CFR 50.54 and what we know about operating large, complex reactor facilities.

CHAIR HALNON: But the other point is, that's the very definition of the entry criteria.

MEMBER KIRCHNER: Yeah.

CHAIR HALNON: So, you're basically saying that any reactor under Part 57 can be GLRO, because you can't get there unless you can meet the 1-rem dose.

MEMBER KIRCHNER: Yeah.

MR. BALAZIK: This is Mike Balazik of the staff.

We wanted to add flexibility in the licensing of operators. So, we didn't want -- what we wanted to do is that, hey, if you have this design where an operator needs to take an action some time down the road, exceed the 1 rem, they would meet that criteria for Part 57. But we didn't want to just throw out any operator actions. So, we wanted to include that as part of below the 1 rem for operator licensing, if that makes sense.

CHAIR HALNON: Well, it makes sense that

you want to build in flexibility, but you basically just said that, if you can get to Part 57, you can be a GLRO. It doesn't matter what methodology you have.

MEMBER KIRCHNER: (Audio malfunction) that it's to operate.

CHAIR HALNON: And so, I can postulate a scenario case and say, under all accident conditions, I never get to half -- I only get to a half rem at my site boundary, which is good, but in order to get that, you know, if I don't trip the reactor within 10 seconds of any transient, I might get to .6. So, that immediate action of tripping your reactor is now under a General License. To me, I don't understand why you even have that. Everything that goes under Part 57 can be GLRO, you need to say that.

MR. BALAZIK: Because we wanted to allow certain technologies, that the operator may have to take an action to keep below the 1 rem. So, for the entry criteria, it can be mitigated. It can be mitigated by safety-related SSCs. It could be mitigated by operator action.

MEMBER KIRCHNER: Wait a minute. Wait a minute.

That's not a passive inherent design, an inherently safe design. That's not a walkaway --

CHAIR HALNON: Yeah, that's --

MEMBER KIRCHNER: -- no operator interaction.

CHAIR HALNON: I've been walking through this thinking that there's this massive --

MEMBER KIRCHNER: That's not consistent.

MR. DICKSON: This is Elijah.

I think some of these things would also be hashed out looking through the accident analysis itself, right? So, when you go and you look at the credible events, we know that the dose analyses themselves are rather quite conservative. There's a lot of margin in these dose calculations.

And when you go through the process of assessing the design itself, if there are things that have to be done in a very quick timeframe, there's going to be training for those operators to do those functions. It can be as simple as filling a tank up over a 5-day, 10-day timeframe.

I mean, those are things that would be assessed in this Chapter 12 analysis. There is a section in the Chapter 12 analysis that looks at human factors. And we would have our human factors team look at those actions and see whether or not those make sense in the consequence analysis itself. I

mean, it is being assessed, if need be, though I highly suspect that these are going to be facilities that don't require operator actions.

CHAIR HALNON: Well, I'm going to have to think through it a little bit more. I'm not real clear on how this limits -- I'm not sure how any microreactor would have a traditional SRO framework.

So, Matt, do you have a comment?

MEMBER SUNSERI: Hey, just the way I'm reading this, Greg -- and maybe you are, too -- I'm just not following it. So, if an applicant meets the entry required for 1 rem dose, if that entry requirement is met because some operator action is taken --

MR. BALAZIK: Uh-hum.

MEMBER SUNSERI: -- then they're going to have to train their operators in the traditional sense.

MR. BALAZIK: Uh-hum.

MEMBER SUNSERI: If they can get to that criteria without operator action, then they can use the General License approach. That's the way I'm reading it. I don't know if that's right or not.

MR. BALAZIK: This is Mike Balazik.

Yeah, that's correct. Again, if you're

crediting an operator action to stay below the 1 rem, we feel that there needs to be a more robust licensing. But if you don't, then you can use the GLROs approach.

MEMBER KIRCHNER: But the problem is that 1 rem is the exclusionary boundary, and that could be huge. We could contaminate a good part of the countryside and not get to 1 rem at the exclusionary boundary. And that takes the trained operator, so to speak, or reduces the expectation on operator response as a defense-in-depth measure to prevent that kind of accident scenario.

CHAIR HALNON: And to be clear, Walt, GLROs are not unlicensed operators. They are licensed operators --

MEMBER KIRCHNER: Right.

CHAIR HALNON: -- but they are trained commensurate to the level of knowledge they need to have.

MEMBER KIRCHNER: Right.

CHAIR HALNON: It's just a reduced commitment to the NRC staff and the oversight, testing, and that sort of -- the program is still approved by the NRC prior to being trained.

So, I'm not attacking the GLRO adequacy.

It's actually the criteria, that it looks to me it's very -- it's almost, I want to say, broad, but it has broader implications.

MEMBER KIRCHNER: Yeah, I'm not attacking the GLRO concept or use of it for, you know, passive inherently safe reactors that have much reduced human requirements in terms of operation, but it just doesn't feel like the right criterion for making a decision on operators.

VICE CHAIR PETTI: We've seen some designs where operator action is required, not necessarily from a safety perspective, but from an investment protection. They have to do something. You know, the analysis is all done the same when they don't do anything; they don't need to do anything.

MEMBER KIRCHNER: Right.

VICE CHAIR PETTI: But from investment protection, they would like them to do stuff. So, there's some gray here that --

MEMBER KIRCHNER: Yeah.

CHAIR HALNON: Okay. Let us just think about this. I mean, in general, the concept is fine. It's just how you implement it and how you get through it, and we're trying to balance the designs that we've looked at and how that would affect -- because they

would fit into the Part 57 world, but --

VICE CHAIR PETTI: I'm more concerned about this, and I wrote a note to myself, because this construct on the surface appears inconsistent with the design attributes. This are passive and inherently safe systems.

CHAIR HALNON: Well, that's where my mind started.

VICE CHAIR PETTI: And that incongruity is what -- yeah.

CHAIR HALNON: Vicki, did you want to say something?

DR. BIER: Yeah. I have a little bit of extraneous question. Do you foresee a nexus between operator-independent facilities and approval or easier approval on how this (audio malfunction)?

MR. BALAZIK: Yeah, I mean, there would be a nexus between the two. I mean, right now, we're working on guidance for remote monitoring and autonomous operation. I mean, that's in development. We actually plan to incorporate that guidance into the Part 57 guidance.

CHAIR HALNON: Okay. Let's think more about this.

MR. DICKSON: I wanted to maybe add, Amy

would like to -- one of our SMEs has her hand up.

Amy?

MS. D'AGOSTINO: Yes. So, I just wanted to jump in and say, first of all, that I don't disagree with anything you all have said.

One clarification that may help here is an operator-dependent facility does have a more -- is required to undergo a more rigorous human factors analysis. So, it brings back in some more requirements that are more akin to what we do for Part 53 applicants; whereas, if it's operator-independent, it is a narrowed scope for the review. So, I just wanted to make that clear, that the operator-dependent would go through a more rigorous review.

VICE CHAIR PETTI: Amy, please identify your last name for the court reporter.

MS. D'AGOSTINO: Oh, sorry. Amy D'Agostino. I'm an NRR human factors team lead.

VICE CHAIR PETTI: Thank you.

Let me just say that we are about an hour and 40 minutes away, and we're not even halfway.

MEMBER ROBERTS: Yeah, Dave, I want to follow up quickly with something Michael just said.

The guidance, the Draft Guidance on Autonomous and Remote Monitoring is kind of skeletal.

It says, basically, you know, bring me a rock and we'll go evaluate it.

But you just said your plan is to elaborate on that as part of the rulemaking. Can you specify?

MR. BALAZIK: Yeah, it is. We have staff that are working on the guidance. They're developing it right now. So, yeah, it's very skeletal right now in the guidance, but we plan to incorporate that guidance within the Part 57 guidance.

MEMBER ROBERTS: Okay. Thanks.

There was also a question in the Draft FRN that would ask for some input on that. Are you going to wait for that as part of your preparing the guidance?

MR. BALAZIK: Yeah. Yeah, definitely. Well, we do have a question to the industry about, you know, their wanting to use remote monitoring and autonomous operation. And we'll take that feedback also to inform our guidance.

MEMBER ROBERTS: Okay. Thanks.

MR. BALAZIK: All right. Next slide, please.

This is the specifically licensed reactor operator requirements. So, these are for designs

where the operator action is needed to satisfy the 1-rem-based entry criteria for Part 57.

And just like Amy said, it requires a more robust licensing approach, more of a traditional Part 55 flavor. It includes a license for individual operators; medical examinations; fitness-for-duty requirements; NRC-approved exams; periodically, qualification, and also, subject to specific conditions within Part 55 and Part 50.

I don't know if there's any questions on specifically licensed reactor operators. Amy mentioned it. Again, more robust licensing of operators approach for these types of facilities.

All right. Next slide, please.

We'll hit fitness for duty at a high level. This slide expands on fitness-for-duty requirements. They're applied to reactors. For reactors that are licensed under Part 57, the approach is intentionally risk-informed and scales requirements based on specific reactor design. Relies on operator action to maintain the 1-rem dose-based entry criteria.

If operator action is not required to keep below the 1-rem dose-based entry criterion, or a credible operator or maintenance errors cannot exceed

the limit, the licensee may implement an FF duty program of its own specification. Again, they have to have one, but it doesn't need to meet the requirements of Part 26.

We have provided high-level guidance for a licensee's FFD program of its own specification. What we did is we listed some core elements, but we do, also, have a question out there for FFD programs of a licensee's own specification on what areas they would want to include in their program.

CHAIR HALNON: So, Michael, this is Greg.

Would you just characterize this as putting more of the fitness for duty in control of the licensee --

MR. BALAZIK: Correct.

CHAIR HALNON: -- as opposed to the NRC?

MR. BALAZIK: Correct.

CHAIR HALNON: So, you're not really saying you can do less. Just do what makes sense, and you bring it to us and we'll look at it.

MR. BALAZIK: For your deployment model business case, do what makes sense.

Okay. So, for operators -- for facilities that do not meet the 1-rem criteria for operator action, we did develop Subpart P within Part 26. And

this is taken after Part 53. What it is, is it's the lowest-tier requirements for Part 53 facilities, and also, the licensee has the ability, if they do not want to use Subpart P, they can also use the traditional approach to Part 26 for the FFD program. And, of course, that's the program for the currently operating fleet.

Okay. Any questions?

(No response.)

All right. Next slide, please.

This is just talking about some of the tools that we have in Subpart P. Again, this is akin to Part 53, where, you know, we have oral fluid and specimen for drug testing, split specimen collections, virtual collections of specimens, portable screening, captured video for behavioral monitoring, and offsite collection.

And this supports, for these types of facilities, that might be deployed very far away. So, licensees can use these tools for their FFD program.

Are there any questions?

(No response.)

All right. Next slide, please.

VICE CHAIR PETTI: Okay. Before you start, we still have a large number of slides and we

still have public comment that we are going to deal with. And knowing what's in our draft letter, most of the stuff on pathways is probably less controversial.

But the Implementation Guidance, it ties so directly to the discussion we had this morning on the design attributes. Could we flip those? Is that a problem? So, we could try to get through that while those discussions are still fresh in our mind? And then, we'll come back to the licensing pathways.

CHAIR HALNON: Thanks.

Justin, did you want to make a comment?

MR. VAZQUEZ: Yes. Because I just wanted to mention, going back to --

CHAIR HALNON: Go ahead and state your name and --

MR. VAZQUEZ: Oh, hi there. Hey, this is Justin Vazquez. I am the Branch Chief for the Access Authorization, Fitness for Duty, and Policy Branch in NSIR.

I just wanted to go back to the Part 26 discussion and just make sure that the Committee had a full understanding as to what we're doing with the new requirements. Because there was a question that came up as to, would the, essentially, self-prescribed programs that these licensees would introduce, would

they necessarily be at the full level of the Part 26 requirements, or could there potentially be a reduction in, like, the level of constraints that are applied by those programs?

And I just want to be clear that the intent of allowing licensees to have a program of their own specification is that they may, indeed, have a program that is more akin to what you would see in other industries. That does not necessarily have the full set of constraints under the NRC's Part 26 requirements.

So, there may be some controls under the NRC programs that a licensee may not incorporate, and that is something that we are trying to give them flexibility with these program requirements. So, I just want to make sure the Committee doesn't walk away with a perception that it's equivalent to a full-blown NRC program. The intent is that it would not be.

They may choose to do something akin a full-blown program or something like Subpart K, but it may be something different from what we've seen before.

CHAIR HALNON: Thank you, Justin.

MR. DICKSON: Okay. With that, we can go to slide 28.

All right. So, slide 28 introduces the Implementation Guidance that will be accompanying Part 57.

The primary document is NUREG-2271. It's a brand-new NUREG. The staff are very proud of putting this together. And it, effectively, covers everything you would need in a submittal for a Part 57 license application.

But it also includes the SRP, the staff guidance, too, as well. So, when you look at the page number and see it's 600 pages, please remember that it's a lot of bulleted, double-spaced items, but then, half of it is intended for the staff to perform their reviews. And it's nice to have it in one document, where applicants can go and see what the staff are going to be asked to look for in the reviews. So, it helps make them produce a better product for us in support of rapid reviews.

The guidance does draw heavily from existing non-power reactor practices, particularly NUREG-1537. It's very similar in structure. It's very similar in content. But we've adapted some of the unique features of microreactor licensing and other reactors of comparable risk profiles for this particular application.

NUREG-1537, I think it was published back in 1995 or 1996. So, we've had a lot of use with it. We know areas that could be improved and we've made those improvements in the document.

VICE CHAIR PETTI: Interestingly, I can remember discussions about 1537 and how old it was.

MR. DICKSON: Right.

VICE CHAIR PETTI: And, gee, don't you think we should dust it off? And here we are.

(Laughter.)

MR. DICKSON: Yeah. Okay.

So, one key aspect, the guidance to the document does incorporate flexible, graded, risk-informed, or hazard analysis approaches to performing the accident analysis that does integrate everything else that you see within the submittal itself.

It does reflect methodologies historically used by the NRC, as I mentioned, NUREG-1537, but also some of the -- a lot of the documentation process for DOE authorization as well. Things like the Handbook on Hazard and Accident Analysis are referenced in this. So, it's a very similar process.

With that, we hope that -- it should have help streamline reviews between DOE authorization, and

then, coming in for a Part 57 license for a commercial application.

So, with that, we can go on to slide 29.

These are all the chapters of the Implementation Guidance. The slide outlines the structure of the Part 57 Implementation Guidance, and it mirrors a full Safety Analysis Report, covering areas such as site characterization, principal design criteria, SSC designation for safety-related or non-safety-related, reactor system descriptions, other safety features, digital I&C, electrical. It covers everything we would need for the review.

The appendices also do provide practical tools, such as source term guidance. There is a deterministic binary SSC classification process there. It is derived off of an older ANSI standard, ANSI Standard 54.1. That was developed in the mid-eighties. It's a binary ANSI standard for large power reactors.

But there are some designers out there that, you know, they don't want to go through a fine-toothed comb categorizing all of their SSCs, because they may not have a lot of SSCs. They're small systems. So, they may choose to have more safety-related systems than they maybe really need for

the purposes of getting these units out the door.

So together, these chapters and the appendices provide a clear, standardized roadmap in developing high-quality Part 57 applications.

I would like to point out that there is Appendix F. This is a graded approach to site characterization for Part 57 applications. This is, effectively, an equivalent of DG-4036, which the Committee had looked at a couple -- I think a couple of months ago.

But due to moving all of these different documents through the review process, we weren't sure if you guys would have looked at that in time. So, we included information specific to microreactors as an appendix in this Implementation Guidance.

All right.

MEMBER HARRINGTON: Elijah, this is Craig.

MR. DICKSON: Yeah?

MEMBER HARRINGTON: The one thing that I guess is not here is how the staff will review for acceptance an application. This is what goes in the application, but the first step is, do I fit under Part 57?

In fact, my earlier comment about how subjective that might be --

MR. DICKSON: Uh-hum.

MEMBER HARRINGTON: -- it seems like that's still an open question of, how do you get to that application acceptance in the first place?

MR. DICKSON: That's a good point. We are encouraging pre-application engagements with the staff, but there's not, like, a checklist that the staff would be following for acceptance review.

Specifically, in the guidance, we do -- Implementation Guidance -- we do suggest to the designers to talk to the staff about their MHA and MCA. And there's tentacles below that. You know, what is the accident analysis process they use to derive those source terms and do we have a good feeling about, you know, their final results. Or hopefully, those types of things get fleshed-out in those pre-application engagement, but, I mean, that's something that we could take back.

VICE CHAIR PETTI: Yeah, I mean, it would be worth maybe having an Item Zero on your list here.

MR. DICKSON: Yeah.

CHAIR HALNON: Yes, this is Greg.

We have seen varying ways people have justified they're under research or test reactor versus not. Having a consistent process to see where

you go, now that we're going to have 53, 57, now 50, 52, there's a lot of variables there having a consistent way to check through that would be a great --

VICE CHAIR PETTI: I suppose that's what you'll pay a licensing manager for, huh?

CHAIR HALNON: Yeah. We're on the record.

(Laughter.)

MR. ARAGUAS: So, this is Christian Araguas.

I'll add, so that's something that I think we're very much looking at, thinking about making sure that we provide clarity to our stakeholders on whether it is better to go 50, 52, 53, or 57. We fully recognize that there's going to be a lot of confusion, I think, from applicants. And I think, as Elijah mentioned, the pre-application is going to be key.

And I think the other aspect, too, I'll highlight, you know, I suspect a lot of advanced reactor applicants are going to want to get into 57. It's going to be our job to really highlight the flexibilities in 53, so that it's not, "Well, gee, I get second prize because I can't get into 57." Right? We know 53 has a lot of flexibilities. We know or we anticipate additional changes through some of the

rulemakings that are likely to come or will come your way. But I think it will become clear that there are pathways that will enable, I think, the timelines that the industry, or particularly applicants, are looking for -- whether it's 53 or 57.

VICE CHAIR PETTI: So, are you guys kind of considering sort of a broader document about the decisionmaking? You know, how do you decide? Or whatever, you know, lineup into the options?

MR. ARAGUAS: We are working on a broader document. I wouldn't call it an SRP or --

VICE CHAIR PETTI: I don't know what to call it.

MR. ARAGUAS: -- a NUREG --

VICE CHAIR PETTI: Right.

MR. ARAGUAS: -- but there is a document that we are working on to make sure that we provide that clarity.

VICE CHAIR PETTI: So, if we put that in our letter, that would probably help?

MR. ARAGUAS: Sure.

VICE CHAIR PETTI: Okay.

MEMBER KIRCHNER: This is Walt Kirchner

Two things that are critical. As we've gotten away from 10 CFR 50, Appendix A, and relied

more on Reg Guide 1.232, if I have the number correct --

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: -- where did the PDCs fit in here?

And then, the other question is, where are quality assurance requirements for both the staff and for the applicant?

MR. DICKSON: We'll get to that in the next slide, I think.

MEMBER KIRCHNER: Okay. I just didn't see it in the --

MR. DICKSON: Yeah. So, quality assurance requirements wound up with operational programs. You'll have a quality assurance program there that they'll have a discussion on.

MEMBER KIRCHNER: I'm concerned more about the design.

MR. DICKSON: That's right. Chapter 3, which we'll go into in-depth, or at least maybe cursory, I suppose, for this meeting, right? We'll talk a little bit about that.

So, we can go on to the next slide.

Chapter 3, this is the chapter that discusses design structures, systems, and components.

And it really lays out how safe the reactor is built. There is a safety-focused design approach here. So, it does emphasize the defense-in-depth, robust safety margins, safety-related features to prevent and mitigate uncontrolled releases of radiological material.

There are advanced reactor expectations. These are the expectations the Commission pointed out in SRM -- in their Advanced Reactor Policy Statement, to incorporate passive inherent safety features, simplified systems, enhanced reliability, and extended operator grace periods.

There is a push to have consistent terminology. So, what the word "passive" is to somebody, I mean, it could mean something completely different to a different designer.

So, we've used both Commission and internationally aligned terms to help everybody get onto the same page in discussing how the reactor operates in regards to these inherently safe characteristics, passive safe systems, failsafe, simplified safety systems. These are all defined in Chapter 3.

We do have an area in Chapter 3, it's a subsection where the applicant will present their

Principal Design Criteria. We do reference, of course, GDC -- oh, sorry -- Appendix A of Part 50, if they wanted to use those GDCs, but we also reference Regulatory Guide 1.232 for the other advanced reactor designs. And there's some high-level guidance there. So, that's where the PDCs would reside.

We also have a discussion tied back to the regulatory requirement itself of 57.30(a) through (f). Those are the Principal Design Criteria attributes; how they are incorporating those design criteria attributes into their PDCs. And then, subsequently, that, then, filters down to how they design the facility. So, those Principal Design Criteria attributes become a very, very important part of designing your PDCs, and then, all of the other systems that follow.

There are discussions in Chapter 3 of how these are hazard-resilient facilities. So, you do look at your environmental events -- right? -- including meteorology, hydrology, the seismic events. All of that is captured in Chapter 3.

And then, there's a section in Chapter 3 that discusses the plant parameter envelope. And so, this is similar to, like, how the AP1000 has a site parameter envelope or plant parameter envelope that

will, effectively, list all of the different design acceptance criterias. Like seismic, wind loads, things of that nature, are found in a table in Chapter 3.

So, for subsequent licensing, what you need to do is make sure that the site that you identify fits within the plant parameter envelope discussed in the FSAR. This is, again, helping with the rapid licensing approach.

Walt, Commissioner Kirchner, Member Kirchner, did I get that all covered?

MEMBER KIRCHNER: I wanted to just add QA to your list. So, is that coming up in another slide?

MR. DICKSON: I'm not talking specifically about QA, but that is covered in -- so, the way we have it set up, from like a large, very high level, is that what you need to do is you need to go identify your safety-related structures, systems, and components. Right?

So, we have a definition that you're focusing on the structures, systems, and components for design basis accidents that need to meet three criteria. They're very similar to the 50.2 criteria for safety-related components.

And you identify these safety-related

components through the accident analysis. Which are these components? What are the structures that you need to keep the doses below that 1 rem, as well as those components that you need to meet those six design criteria attributes?

From there, you would then go look at what standards are necessary to design those particular components. And so, that's kind of like, I guess, a fault tree, I suppose, of how that's supposed to be done. It's a similar process to NUREG-1537.

MEMBER KIRCHNER: There would be an overall requirement, either at the manufacturing license site or elsewhere, for a QA program. Right?

MR. DICKSON: That's right.

MEMBER KIRCHNER: Yeah.

MR. DICKSON: Yes.

The requirement for codes and standards can be found in 50.60(a)(9).

MEMBER KIRCHNER: Yeah.

MR. DICKSON: Okay. All right. We'll go on to slide 31.

So, Chapter 12 really is the heart of how an applicant demonstrates safety under Part 57. It is a consequence-oriented framework. Part 57 emphasizes outcomes by requiring applicants to establish the

licensing basis through Chapter 12 accident analysis, ensuring that performance is demonstrated rather than prescribing specific design features. This is being done, basically, in direct response to the Commission direction in SRM-SECY-98-144.

It is a graded, systematic risk evaluation approach. The rule adopts several different methods in regards to -- well, it's very open to how you would perform the accident analysis. You can use any one type of risk assessment method to categorize your SSCs, but you just need to present us to what that method is -- trying to be more in line with a 1537 or DOE authorization process when you do your graded hazard analyses through those frameworks.

There is a requirement for an applicant to define or to present to us what their risk assessment method is. They can find that in 57.60(a)(12). So, even before we go and look at the MHA, one would want to go look at this requirement to understand what's the method they're using to develop their MHA or their MCA. So, there will be discussions in the FSAR to help support those reviews.

And it still follows a very traditional nuclear safety framework. We are still relying on defense-in-depth, large safety margins, and focusing

specifically on safety-related SSCs for managing the design basis accidents.

In Part 50, it's a bit broader, or much broader, where you look at design basis events. We've narrowed it down to design basis accidents specifically.

CHAIR HALNON: Elijah, you mentioned that, under the flexible risk analysis, the MHA and MCA. My mind went to more deterministic on those accidents.

MR. DICKSON: Yeah.

CHAIR HALNON: Is that something different? Are you actually using risk metrics in determining MHA?

MR. DICKSON: No, we're not using -- it's the process of how you gain confidence in what that MHA is.

CHAIR HALNON: Okay. The end point?

MR. DICKSON: The end point.

CHAIR HALNON: Okay.

MR. DICKSON: That's right.

We can go on to slide 32.

VICE CHAIR PETTI: Elijah, you know, I looked at this and I see -- I mean, it looks an awful lot like 53.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: And maybe the devil is in the detail in terms of the flexible risk analysis.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: But, you know, the final Part 53 language went from PRA to a broader risk term, and I can't remember the exact --

MR. DICKSON: Yeah, systematic risk evaluation --

VICE CHAIR PETTI: Right.

MR. DICKSON: -- I believe is what it is. And this is --

VICE CHAIR PETTI: That's not that different than what's here.

MR. DICKSON: Uh-hum. This is a systematic risk evaluation.

VICE CHAIR PETTI: It is. And I can remember in discussions pointing to the DOE process, which you guys have brought in --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- in 1537, I think, as well.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: Then, the more qualitative risk.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: And we thought that, for some of the lower hazards, this would be exactly what be needed, would be really useful.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: So, I'm happy to see it.

MR. DICKSON: Okay.

VICE CHAIR PETTI: Again, this is where it does look like it's blurring. And I just suspect the applicants who have a new, really new --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- you know, are not going to understand the nuances of this.

MR. DICKSON: Yeah. I mean, applicants that are very new, looking at an LMP approach or a simple design, I think would struggle, utilizing that type of an approach for a simple design.

If you go to slide 32, we can talk about how one goes about assessing -- did you want to say something?

MR. ARAGUAS: Let me pause there. Are we going to 11:15? Or when is our cutoff? Because I want to make sure, if we are, we just cut this --

CHAIR HALNON: Go through this slide, and then, we'll stop after this slide.

MR. ARAGUAS: Okay.

CHAIR HALNON: Because I assume there will be a lot of discussion on the next set of slides.

MR. DICKSON: Uh-hum.

MR. ARAGUAS: Perfect.

CHAIR HALNON: Yeah, so go ahead and get through this one.

MR. DICKSON: Okay. Okay. So, these are all the different subchapters of Chapter 12, of NUREG-2271, proposed NUREG-2271.

And the idea is that we need to have confidence, the staff need to have confidence, in the uncertainty of their proposed MHA or MCA source term, which then, subsequently, defines where their site boundary is at.

And so, we have this. I think it's a pretty nice figure that shows how you go about that process. And you don't need to use PRA, traditional PRA, to go about doing this. I mean, you can use HAZOP. You can use FMEA. You can use event trees. You can use fault trees. You can use joint event trees and fault trees to understand how your system interacts to the different types of events that you expect at a facility.

And so, from there, what you do is you

look at all of your fault -- you postulate all the different types of events that can occur. And you categorize these events into different categories of similar plant responses, or maybe initiators, external events. You can have an external event category. You can have a reactivity insertion-type category. These are all very similar DBAs that you would see in NUREG-1537.

So, from there, you have these bins of design basis events, and then, within each of those design basis events, you would assess the most consequential events, and you would label those as design basis accidents for that category. This is exactly how we, effectively, do it under a Part 50 license.

Then, from there, you look for the most consequential design basis accident, and then, you derive your MHA or your MCA. So, this is a logical process in regards to assessing how your facility responds to off-normal events.

And within this guidance chapter, we tell the staff and the applicants to address, you know, what is the methodology you're using? Are you using HAZOP? Are you using FMEA? Are you using fault tree/event tree? Are you using LMP? You could use

LMP is this process.

And then, from there, we go down the process of narrowing down what are the most consequential accidents we need to be looking at.

VICE CHAIR PETTI: So, let me just ask a question. Because there is some discussion about -- you know, in Part 50, there's DBAs, and then, there's that handful of BDBAs that are in law --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- the station blackout, ATWS.

The rule, because you picked up 1537, talks a lot about loss of offsite power.

MR. DICKSON: Right.

VICE CHAIR PETTI: But it's really silent on ATWS. And as I understand it, ATWS in test reactor space is generally not looked at. But when I read the words, it seems to imply you're looking across everything, and you look at where your BDBE -- you know, for frequency space, ATWS is there.

And what Tom and I are concerned about is that there's a lot of reactivity being held down in a microreactor.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: And there's not enough

space for the rods. So, their worths are very hot.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: And so, would ATWS be considered as an event here in the guidance? Because it's kind of silent.

MR. DICKSON: I can discuss --

VICE CHAIR PETTI: I can interpolate and interpret, but --

MR. DICKSON: Yeah, it does not specifically assess ATWS. It's not something that we've typically done for NPUF-type design.

Going back to the design criteria attributes, though, I mean, the decay heat, action for the heat removal attribute does say that you need to cool the reactor down passively without power. You would assume that you get rod insertion, right?

VICE CHAIR PETTI: Right.

MR. DICKSON: But, you know, there's several. There are several severe accident regulations. Well, we've got station blackout. We do have 50.67 for, you know, the accident source term. Those are all regulations that were put into place after the fact, that we identified an issue.

VICE CHAIR PETTI: Right.

MR. DICKSON: And then, the Commission

went and took action to make sure that it's addressed.

And I can see, you know, if something were to occur, those same actions would be followed by the Commission; that we would have -- but hopefully, those types of things -- oh, I'm sorry.

VICE CHAIR PETTI: But this process, I read it and I go, yeah, you need to look at everything.

MR. DICKSON: Yeah.

VICE CHAIR PETTI: And ATWS would definitely fall into the BDBE bucket --

MR. DICKSON: BDBE.

VICE CHAIR PETTI: -- as you have described it.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: But there's no words about it. Because that's what station blackout will be, and that's mentioned, you know.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: So again, there's this incongruity because, in 1537, you haven't historically looked at it. But further, this is the weakness of a microreactor.

MEMBER KIRCHNER: Yeah, this is a --

VICE CHAIR PETTI: This is a fundamental

weakness, and you need to assure yourself that you know that there's high reliability that that system is going to shut down.

I mean, there are these micros that they really reduce the power density. And so, the burnup is the life of the core; a 30-year reactor and never have to change the fuel. Do you know how much reactivity they're holding down? You know, multi-dollars.

MR. DICKSON: That design criteria attribute reactivity was intended to help hold down stuff in the reactor full of excess reactivity.

VICE CHAIR PETTI: Right, but there's not enough -- there's not enough physical space to spread that reactivity across enough rods, so that in the ejection it's not a safety net. We're talking -- it could be, I mean, we've seen designs --

MR. DICKSON: Yeah.

VICE CHAIR PETTI: -- it can be a very serious, serious event.

MR. DICKSON: There's --

VICE CHAIR PETTI: That's all. We're trying to make sure people focus on that. Because, again, this is something that you wouldn't, if you're thinking of the big ones, it's a completely different

story.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: As you make them smaller, this is one of the constraints that comes up, and it's an important safety issue.

MR. DICKSON: Yeah. Chapter 12 does have a DBA category for reactivity, and it talks about looking for -- doing a rod ejection type of analysis on the most reactive location in the core.

MEMBER KIRCHNER: Well, I can tell you going in, if you have that kind of an accident, you're in severe accident space --

MR. DICKSON: You're right.

MEMBER KIRCHNER: -- with these micros, because of the reasons that Dave explained.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: The rod worth or drum worth is so much more in a smaller system than it is in one of the large reactors. So, a PWR can survive a rod ejection accident, while these smaller ones are not going to.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: So, if that's in the MHA or MCA -- now, I don't know how the applicants are going to approach this, but, to me --

MR. DICKSON: Yeah.

MEMBER KIRCHNER: -- that's a credible accident.

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: I suspect they will not want to analyze that.

VICE CHAIR PETTI: It will affect the entire core.

MEMBER KIRCHNER: And they have a design basis --

VICE CHAIR PETTI: It's not like in a light reactor. It's just a low (audio malfunction) --

MR. DICKSON: Right.

VICE CHAIR PETTI: -- I mean, because these, they're small.

MR. DICKSON: Especially the fast reactor designs are highly --

VICE CHAIR PETTI: Oh, I will tell you that, even a small TRISO --

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: -- we're talking about -- you talk about the temperature limits we have. That's all out the window.

CHAIR HALNON: Okay. On that happy note, we're going to --

(Laughter.)

MEMBER MARTIN: Actually, I have -- I would like to jump on this as well.

CHAIR HALNON: No, you cannot. I'm sorry, we're out of time. We can pick this back up after lunch. But I have to stop it because we've got a hard stop in a few minutes. And I do have to make some announcements.

This is a reminder that the ACRS has some internal activity beginning very shortly and we'll be going until one o'clock.

The public comment portion will be after we continue, finish with the slides after lunch. And we're going to try to limit that to one and a half hours, so that we can proceed with our deliberations and letter writing.

And we're going to ask the commenters to limit their remarks to three minutes or less. And if we have additional time, then we'll allow them to expand their comment after everyone's had their opportunity.

I do want to clarify that or restate that the public comments are solely for the benefit of ACRS deliberations and they will not be considered as part of the rulemaking docket. If you want to make your

comments on the rule in the docket, then you have to go to regulations.gov to make that comment.

Also, if you wish to make a public comment -- because we're anticipating there may be a lot -- you can secure your place in line by emailing Larry Burkhart at lawrence.burkhart@nrc.com and Weidong Wang, weidong.wang@nrc.com.

Did I say "com"? I'm reading "gov."

(Laughter.)

I'm sorry, it's .gov, nrc.gov.

And Weidong is W-E-I-D-O-N-G, dot, W-A-N-G, @nrc.gov.

We're going to create the list of speakers in line, so that we can make sure that those folks that have the comments can make them. It's not actually necessary; we'll take comments as we go, but we'll try to keep some structure to it.

And then, again, a reminder that, on May 20th, the staff will be holding a public meeting for another comment -- or another opportunity for the public comment.

So, with that -- it is 11:23 -- we're going to be in recess until one o'clock, and then, we'll continue on with this slide right here, finish up the comments. Bob's first.

MEMBER MARTIN: Thank you.

CHAIR HALNON: Okay. So, we're in recess at this point.

(Whereupon, at 11:23 a.m., the foregoing matter went off the record for lunch and went back on the record at 1:00 p.m.)

CHAIR HALNON: Good afternoon. This is the 735th meeting of the full Committee of the ACRS. We're now back in session for our afternoon session.

As I mentioned before we recessed, we are taking email comments, or at least getting in line through email from the public comment perspective. We've gotten a few people in line. So, as we go through the rest of the slides, if you want to make a public comment and you want to get in line, please email Larry Burkhart or Lawrence Burkhart or Weidong Wang. Those email addresses are in the chat.

In addition to that, just to let you know that it doesn't look like we're getting a lot of people in line. So, there will be probably time for expanded public comments, but we'll take about an hour, hour and a half of comments.

So, at this point, I'm going to turn it back over to Christian to get the afternoon session going.

MR. ARAGUAS: Sure.

MEMBER MARTIN: Was I going to get my question in?

CHAIR HALNON: Yeah, but we're not -- he wants to circle back on another topic first, and then, you can --

MEMBER MARTIN: We're coming back to this?

MR. ARAGUAS: Yeah. Sorry, if you don't --

CHAIR HALNON: We're going to go back to operator licensing to go ahead and clarify some things, and then, let you open this session back up with your comments.

MR. ARAGUAS: And I appreciate you reminding me.

MEMBER MARTIN: Yeah. And you can count on me.

(Laughter.)

MR. ARAGUAS: And I will try to be quick, so we can get to your question.

MEMBER MARTIN: All right. All right. All right.

MR. ARAGUAS: So, this is Christian Araguas with the NRC.

And I wanted to go back to the discussion

we were having on operator licensing. I think there was questions around the framework in terms of how you meet the 1 rem at site boundary entry criteria, and why wouldn't you just impose -- and I'm putting words in there -- why wouldn't you just have the GLRO framework? Why do you have a situation where you're defining an operator-dependent facility which relies on, you know, operator action to meet the 1 rem at site boundary?

And so, I wanted to go back and just highlight that the rule -- you know, we have the design criteria. They're there to strongly encourage the reliance and use of passive features, but they don't require that a facility can't use active systems to achieve that 1 rem at the site boundary.

So, because we're not prescriptive, it's performance-based, I think the idea is you could have an applicant that could rely on active systems to achieve that entry criteria. And so, because of that, we want to make sure that we have a flexible framework for how you address operator licensing.

I think we fully expect to see the majority of the designs are inherently safe, relying on passive features. But because the rule doesn't preclude the use of active systems, we want to provide

these flexibilities in the rule.

VICE CHAIR PETTI: Well, okay, I get that. But the matter of construct, because whether it's in the FRN or -- there is a construct around simplicity. And if you've got active systems, that tends to go the opposite direction of the passive and inherent system.

So, it may require some tweaking of language to kind of be more internally self-consistent.

MR. ARAGUAS: And we'll take that comment. Thank you.

CHAIR HALNON: Okay. Is that where you wanted to go?

VICE CHAIR PETTI: Yes.

All right. I have to call on Bob to ask his question.

(Laughter.)

He's had too much time period to think about it.

MEMBER MARTIN: Yeah, too much time. Too much time.

Where we left off in the morning, Dave -- and I guess influenced by conversation with Tom -- related to a specific event, ATWS. I think you also acknowledged that Part 50, you know, despite all the

discussion on design basis, really has a lot of beyond-design-basis elements in it.

In Part 57, and probably in Part 53, and really anywhere that we're discussing risk-informed in the context of an application of LMP, which, effectively, if not implicitly -- it's explicit that the mechanistic-source-term approach is just part of LMP, or required even.

Elijah, of course, you made a special note. You're not prescribing a particular evaluation model. However, a mechanistic-source-term evaluation of a maximum credible accident is going to give you a dose that is, could be two orders of magnitude lower than a traditional alternative source term MHA.

MR. DICKSON: Uh-hum.

MEMBER MARTIN: And now, yeah, you might say the onus is on the applicant to justify. But, you know, I went back to the SECY which I think you were referring to, which is 93-092, which there was a lot going on in '93, you know. But it, specifically, is the document that prepped the Commission on the Advanced Reactor Policy Statement.

MR. DICKSON: Uh-hum.

MEMBER MARTIN: And MST is clearly a best-estimate methodology, right? A best estimate of

uncertainty.

But those models went towards or are best applied when you have aleatory knowledge, and that uncertainty could be well-characterized. The historic argument, you know, historic argument is applied across accident domains, that we have conservative models to address epistemic concerns.

Now, it's a fair argument that we know a lot more today, but the pivot from a methodology that you might say, given our new knowledge, might be 50 to 100 times more than reality.

And I'll give, for example, Fukushima. Right? This is a plant we have data on. Three plants, of course, had core melts. And the dose at the boundary was a few rem, in that order. Okay. If you had a mechanistic source term, you might predict a few rem. Or if one plant, it might have been 1 rem. Right?

So, we all feel pretty good about, you know, how the containment held up. I mean, maybe we don't feel real good about it, but, you know, it could have been a lot worse, you know, in another circumstance.

But, nonetheless, you start scratching your head going, well, is 1 rem the right thing? Now,

that opens up a whole can of worms. I'm sure you've already kind of gone on that.

One characteristic of LMP -- and, you know, picking on something that you might say we've already talked about and come to some settlement, it is inappropriate. But as an independent member of this independent body, it's not exactly agnostic in how you approach safety analysis -- meaning that, you know, clearly delineates design basis and beyond design basis when it comes to preparing deterministic safety analysis. It adopts the old Part 50, where, okay, we can do best estimate. In fact, you can do a lot of screening in this space; we forget about it, right?

Kind of contrary to risk-informing, where you're really supposed to blur the lines. Now, it does blur the lines to some extent. And again, it's the same kind of thing. Somebody, one applicant could come in and say, "Yeah, we're acting on the whole space; we're doing it." And another one could come in and maybe make some arguments and they start filtering out that kind of opportunity. It's not objective. And so often, the arguments, the justifications become qualitative, particularly with non-light water reactors because we don't have data, and we don't have

data that is epistemic, and certainly is huge.

So, how do you justify allowing almost like complete elimination of epistemic uncertainties by saying, yeah, you can go all the way to the extreme of MST on MCA, and still acknowledge that there's value in AST and MHA?

To me, I guess, at the minimum, you should take a look back at what we are saying applicants can do and be upfront, explicit. Epistemic uncertainty is there in all these plants, and at least there's a factor of safety. Maybe it's not 100 in using MHA and AST, but it's something.

And you can't just allow -- it's not the equivalence of safety. Remember, that's the big part of the Advanced Reactor Policy Statement -- some kind of equivalency of the application of -- maybe they don't use the word "Part 50," but they'll use a word like, you know, "the attrition of LWR safety."

CHAIR HALNON: Is there a question in there?

MEMBER MARTIN: I just said it, about the epistemic uncertainty and the use of MST and MCA, and how different it is, particularly and, of course, in light of the same acceptance criteria.

CHAIR HALNON: I didn't hear a question.

I heard a preach, but I didn't hear a question.

MEMBER MARTIN: Well, I said, this is my question. Well, I don't know. It will go up in the transcript. But anyway, it was related to the epistemic uncertainty is not being treated. It's a complete -- it's open in Part 57; it's open in Part 53, and it's open everywhere, honestly, wherever MST is being allowed to exist, which is, ultimately, at LMP.

There's a lot of acronyms in all this. I realize it gets hard to --

MEMBER KIRCHNER: I think what Bob is trying to say is, why not just use an MHA as the bounding criterion?

MEMBER MARTIN: Well, or alternatives. Maybe we know enough to have an alternative, but I'm not sure -- when you use a number like 1 rem, it just means something totally different in the whole spectrum of evaluation models.

But by saying you can do all of that, you're telling the industry that, eh, we don't believe epistemic uncertainty is that important -- in a time where none of these plants exist; there is no data. We're going to allow carte blanche on these kind of evaluations?

And even if you say we'll scrutinize that and make sure of the justification, you're inviting them to spend a lot of time, potentially, wasting their time, on an evaluation model that can't possibly address the epistemic uncertainty.

So now, this gets to something that might get addressed in a Reg Guide, even the NUREG. But I'm uncomfortable with 1 rem when you, basically, say beyond-design-basis accidents, where dose comes from. I mean, where what really matters to safety comes from is being treated differently than design basis accidents. It should be agnostic across the board up until whatever we decide is low enough. You know, is it 5 times 10 to the minus 7? But we should have the same kind of deterministic analyses that backstop the envelope for these designs.

What we'll find out is that these doses -- use yesterday, for example, or use Kairos last year, or maybe even TerraPower -- these doses are going to be well below 1 if they're using well below 1.

And what's the point of all this exercise if you have two orders of magnitude of margin? Because, then, the argument is like, well, why are we wasting our time even to go to the NRC? We've got two orders of magnitude on a dose.

So, there's a lot there, but you gave me an a hour and a half to think about this.

(Laughter.)

CHAIR HALNON: I gave the public three minutes apiece, and you've taken nine. So, I don't know what to say about that.

MEMBER MARTIN: But I think there's structural, something structural wrong here on the freedom that is apparent in the way it's written. And there are no guardrails, as far as I'm concerned, with the scope of the evaluation models.

MR. DICKSON: Okay.

MEMBER MARTIN: Because you have both extremes explicit in the current ruling.

MR. DICKSON: Yeah, this is Elijah with the staff.

The ruling, which doesn't prescribe a method --

MEMBER MARTIN: But it does acknowledge MST and MCA --

MR. DICKSON: That's correct.

MEMBER MARTIN: -- and MHA, but it doesn't say AST. MHA, basically, comes along with AST.

MR. DICKSON: Right. We still have defense-in-depth practices, right?

So, you go back to the Part 50 licensing framework. We did not have the calculational tools that we have now to be able to assess some of the things that I think you're talking about in regards to -- we'll talk about TMI.

You know, we didn't have those types of PRA tools to assess those accident sequences, but we did identify the TMI accident. It's the WASH-1400 report or the Rogovin Report? Yeah.

With the tools that we have now, we were able to, with Sandia National Lab, do forensics work and do a very good job of modeling the events that caused the core melt. And those offsite dose calculations that we can do with the MACCS Code 2 do a pretty good job. We have much better tools now to do these assessments than we did back 70 years ago.

So, going back to, like, the Part 50 model, where we were using an MHA, that was a source term that was -- it just so happened to be belched into the containment. Right?

MEMBER MARTIN: Right, right.

MR. DICKSON: And the purpose of that source term was to assess containment function. And we always assumed that the containment always functions in that analysis. It didn't leak past the

tech spec leak rate values.

Even though we know, if you had a beyond-design-basis-type LOCA event, you could reach containment.

MEMBER MARTIN: Right, right.

MR. DICKSON: Right? And that did happen.

MEMBER MARTIN: Right. But this is kind of apples and oranges. Yeah, we're trying to sharpen the pencil, in a way saying we have really good knowledge. We're not going to say we have perfect knowledge.

MR. DICKSON: Uh-hum.

MEMBER MARTIN: But I'll go along with the fact that we have really good knowledge. But we have analogies -- here is the analogy: we have really good knowledge about structures.

MR. DICKSON: Yeah.

MEMBER MARTIN: Right? And ASME Boiler and Pressure Vessel Code still has, like, a factor of four --

MR. DICKSON: Uh-hum.

MEMBER MARTIN: -- on, you know, with its design loads.

MR. DICKSON: Uh-hum.

MEMBER MARTIN: You know. You would think

that, at least on something like dose, we would at least have acknowledged the need for a factor of safety. It doesn't need to be something ridiculous like the old methods, given that we know a lot more today. But to basically allow best estimate, some best-estimate judgment of what's credible --

MR. DICKSON: Uh-hum.

MEMBER MARTIN: -- seems to be a bridge too far.

MR. DICKSON: Okay.

MEMBER MARTIN: You know?

VICE CHAIR PETTI: Well, but the guide talks about safety management. It's not like they're jettisoning the safety margin. They're expecting safety.

MR. DICKSON: There's a whole section in the guide for a defense-in-depth evaluation.

MEMBER MARTIN: Like I said, even if it gets judged through a review process, if somebody comes in and looks at it, basically, it's explicit. You could come in with MST with a MCA strategy. I mean, they could be just wasting their time until, you know, the smart guys that are reviewing this thing come back and say, "Aw, well, you just don't have the justification." Right?

But it also brings into your point, where does ATWS live? Where does SBO live? Where does LOCA live? You know, these things live in that beyond-design-basis space, and I think they need to continue to live in this space --

MR. DICKSON: Even for --

MEMBER MARTIN: -- for microreactors.

MR. DICKSON: For micro --

MEMBER MARTIN: Because, again, that's where your source term is really going to come from.

MR. DICKSON: Uh-hum.

MEMBER MARTIN: And it is, in my mind, equivalent to the level of safety that we expect with LWRs.

All right. Rant over.

MEMBER ROBERTS: You know, I think it might be good for Elijah to move on to the next slide, because a lot of this is on the next slide.

Because my internalization about what you just said is that they have epistemic uncertainty associated with the MCA. It is hard to quantify because somebody has made a judgment: this is credible.

MEMBER MARTIN: Yeah.

MEMBER ROBERTS: Which means they've made

a judgment this other thing is not credible. And the epistemic uncertainty you're talking about, I think, is what's deemed not credible, because somebody didn't think of it, or somebody underestimated the likelihood, something like that.

MEMBER MARTIN: Both spaces have, you know --

MEMBER ROBERTS: And so, if you have one of those scenarios, then the MCA is potentially not conservative because it missed some important phenomenon. Actually, the MHA would cover that. And I think your next slide is probably a good thing --

MR. DICKSON: The next talks, I think, a lot --

MEMBER ROBERTS: -- because it talks about the two and it compares them and how you might use one and not the other.

MEMBER MARTIN: Right.

MEMBER ROBERTS: So, it's my suggestion to move on.

MR. DICKSON: All right. So, on the next slide, I think this is the fun part of the presentation really, quite frankly. Everybody has thoughts and feelings on MHA and MCA.

(Laughter.)

MEMBER MARTIN: I'm not emotional.

(Laughter.)

MR. DICKSON: I have been looking forward to this part of the presentation for -- I don't know -- several months now.

So, when we first started working on Part 57, I had asked some folks, what are some of the things that we need to be kind of bringing along with us from Part 50 or 70 years' worth of licensing? And one thing I've always felt is that we've never had a good definition of what an MHA or an MCA is.

And so, they said, well, you have a clean sheet and you can go ahead, and for the purposes of microreactor licensing, let's start fresh, so we can get everybody on the same page as to what the MHA and what the MCA is; what their purposes are. And this is discussed pretty clearly, I think, in the Regulatory Guidance 2271.

So, slide 33 focuses on the two primary bounding accident methodologies using Part 57. It's a maximum hypothetical accident and a maximum credible accident. These are essential for determining the site boundary to demonstrate site boundary location, to demonstrate the 1-rem acceptance value. It's also important for designing and sizing mitigative systems,

if you need such a system. And this is the 1-rem criteria in 57.25(a).

So, the MHA and MCA analyses, they are intentionally conservative. You typically assume a large source term, potentially, with significant core damage. You assume maximum leak paths from the core to the containment or confinement, and then, out to the environment. You assume adverse weather conditions. You can think of your Sunday morning-type weather conditions. Typically, that's 95th percentile meteorology that we're using. And that the individual is located at the centerline of a plume with no protective actions. So, this is a very conservative dose analysis. This is, you know, after you come up with your MHA or MCA source term.

So, the assumption is that we're ensuring, even under the most unfavorable conditions, or very close to it, that we're still meeting the dose acceptance criteria in any given day of the year.

So, what should the source term or what does it must include? We have defined what the source term is in 73.3. This is typical -- it's like a Reg Guide 1.183 analysis for Part 50 reactors.

Radionuclide inventory, physical and chemical form, release fractions as a function of core

inventory, and then, those release timings. This would be, in effect, your mechanistic source term that you're referring to.

So now, if we go to the left side of slide 33, we're discussing what the MHA and the MCA are. And so, the MHA is certainly the more conservative of the two approaches. And when I get done talking about the MHA and MCA, I'll give you some examples of what I've come across in FSARs for facilities that have done either/or approach or both approaches.

So, it assumes a severe radiological release that is physically plausible, but it does not have to represent a realistic or credible sequence of events. It's a highly bounding scenario, useful when applicants want to avoid detailed mechanistic modeling or when some system performance under extreme conditions is uncertain.

The maximum credible accident, in contrast, the MCA, while it's still conservative, it focuses on events that are credible, given the class technology, the safety features, and operational characteristics of that facility. The MCA does avoid physically unrealistic assumptions, and instead, represents the most severe radiological release that could credibly occur.

The primary difference between the two is the magnitude of the source term, which impacts several downstream design and siting considerations; specifically, the site boundary, the number and type of designated safety-related pieces of equipment, which then filters into how much quality you need to assign to these pieces of equipment, and the need for mitigative systems.

In Part 57, both approaches are acceptable, and the guidance recommends that the applicants do start pre-application engagements with the staff in regards to what they're going to propose for their MCA or MHA.

Now, I went and was looking back at some of the NPUF -- I think it's Chapter 13 -- hazard analysis reports to get an idea of what some outfits have used for the licensing. And I found one, in particular, that was striking, because it did provide, like, a good visual description of what the MHA is and not some hypothetical accident.

And they also did an MCA, and they did present both the doses for those. And so, the MCA -- we'll start with that one -- was, in effect, a control rod ejection type of accident. This is for, I think, a TRIGA reactor, is what it was, a 1-megawatt type of

reactor. And then, they did the dose calculation and they were really very low at the site boundary.

And then, they did the MHA, and the MHA was they assumed, they hypothesized that the reactor exploded and that 100 percent of the fission products went volatile, and then, performed their dose analyses.

And I would encourage anybody to go to their favorite, like, ChatGPT, and type in what an MHA versus an MCA is. You'll get some great pictures, visual pictures, of what these things are.

And so, that's an example of an MHA that was used. And for this, like, 1-megawatt, TRIGA-type reactor, they found that, at about 800 meters, that's where they would not receive lethal doses, and whether or not they needed to do emergency planning-type actions. So, this is something that was done back in the fifties.

So again, just trying to paint a picture of what these two concepts are and how they've been used in the past.

In the guidance document itself, we do discuss MHAs and MCAs for several different reactor types. Went and pulled -- did a literature review, found the information to give applicants and designers

an idea of what they could be proposing to the staff, and also, to educate the staff in the future as to what these are.

So, with that, any questions on slide 33?

DR. BIER: Vicki Bier.

In a way, it is partially going back to a previous slide, but I think one of the things that was problematic in the early days of defining maximum credible accident is that, before WASH-1400, we didn't have a way to judge credibility, because we didn't have the probability side.

And if you are allowing qualitative methods like FMEA and HAZOPs, then does that interfere with being able to judge credibility or are you defining credibility kind of more mechanistically in a sense?

MR. DICKSON: Well, I think it goes back to the previous slide, how you start with whatever could happen at the facility. You get a bunch of engineers and scientists, operators in the facility. If you're doing a HAZOP type of analysis -- you've got to start somewhere. And you postulate all these different events that can occur, and then, the way the guidance has it set up is there's another systematic risk evaluation. And you categorize and bin -- it

could be several thousand events. Bin external events into one. It's like events that impact a facility. That's what we're proposing in this rule.

And I don't think you necessarily need, you know, failure data to help inform credibility when you're looking at perhaps these very small microreactor designers where they may be relying on gravity to effectively shut the reactor down, and it can handle the decay heat passively.

I mean, we're trying to really focus on those, but it does get more complicated when the reactors get larger.

DR. BIER: So, you're saying that somebody could define an MCA without doing a probabilistic analysis?

MR. DICKSON: Yeah. Yeah, that's right.

DR. BIER: Okay.

MR. DICKSON: Yeah. An example being, you can do a fault tree/event tree analysis without tacking on the probability numbers and still assess how different deterministically different pathways to failure of that core.

MEMBER ROBERTS: How are, like, ATWS events treated under this construct? Is a failure of the reactor protective system consider credible? If

you do a PRA, you'll come up with some probability probably pretty low.

MR. DICKSON: Right. I would think that you would still be applying your same type of safety analysis concept. So, if it's a safety-related system of high quality, then you would credit that to a performance function in any one of these credible events.

MEMBER ROBERTS: Yeah, I'm trying to understand how that aligns with, say, the existing 10 CFR 50 approach, where the assumption is that the emergency core cooling system does not function. And so, there's a safety-related system. To get your source term, there's a whole sequence of interactions.

MR. DICKSON: Oh.

MEMBER ROBERTS: That's one of them would be considered its own credit. You know, the system there, is that system error not credible?

MR. DICKSON: Yeah. So, okay, you're referring to 50.46 and the ECCS rule. Is that the --

MEMBER ROBERTS: Well, 50.34 and the source term that's used for 50.34 for the 25-rem --

MR. DICKSON: Yeah.

MEMBER ROBERTS: So, that's not a crediting safety system. That's a postulated --

MR. DICKSON: That's a postulated source term, that's correct.

MEMBER ROBERTS: It's closer to MHA.

MR. DICKSON: It's --

MEMBER ROBERTS: In this case, if you're using an MCA, and if, like we talked about earlier, if there's so much excess reactivity held down by the rods --

MR. DICKSON: Uh-hum.

MEMBER ROBERTS: -- if you let the rods go out without a scram, you would have a really bad day, not bounded by 1 rem.

MR. DICKSON: Right. It wasn't assessed.

MEMBER ROBERTS: Is that credible or are you allowed to say, "Well, I've got such a good reactor protective system that it's not credible."?

MR. DICKSON: Uh-hum.

MEMBER ROBERTS: So, which are you thinking would be the approach you would expect?

MR. DICKSON: I would expect an MCA type of approach, where they would assess whether or not that's something that needs to be credited in the dose analysis itself. It would be. I would think it would be. But I don't have a good answer for you on that one, you know.

VICE CHAIR PETTI: Again, this is, again, this issue picking up stuff from 1537. The loss of offsite power is there in the words. You don't say, "station blackout," but the words are there, because that's a 1537 thing. So, you can see a parallel to what's in Part 50, right?

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: Then, ATWS is over there, but it's silent over here.

MR. DICKSON: Right.

VICE CHAIR PETTI: Although, as I read how you're supposed to get there for MCA, you're supposed to look at range of events, your table, your cables with risk.

MR. DICKSON: Right.

VICE CHAIR PETTI: It's there. It's got to be in that, unless you've got some amazing reactor protection system.

MR. DICKSON: Right.

VICE CHAIR PETTI: Which is really all I'm asking is, they had better make sure they have a stout reactor protection system, because that event is particularly bad. Whereas, in common plants, they has been designed away mostly.

MR. DICKSON: Uh-hum, uh-hum.

VICE CHAIR PETTI: They don't have the flexibility in the design space, as many options. So, it's really going to come down to RPS and making sure that it's got everything, every bell and whistle it can have.

MEMBER ROBERTS: So, you, basically, need a system design that supports the credibility assumption. Whether that's good enough to have a single reactor system or whether it requires two layers of diversity or three layers of diversity --

MR. DICKSON: Uh-hum, uh-hum.

MEMBER ROBERTS: -- whatever is required will come out of some other evaluation.

MR. DICKSON: Yeah.

MEMBER ROBERTS: And I'm wondering if that evaluation would be within the bounds of Part 57, whether that's the kind of thing, you know, that will kick you into Part 53 because of the need to consider --

MR. DICKSON: Yeah, I --

MEMBER ROBERTS: -- to many more details, what kind of features you need.

You know, that's really the key here, is that, in either case, you would have to come out the other end knowing that you really think that's a very,

very low-frequency event.

But is this the right way to get there? Or is that part of the --

MR. DICKSON: I think that's something that would be assessed through the evaluation process itself. Again, this goes back to how the guidance is set up, as opposed to having a requirement --

VICE CHAIR PETTI: It's not precluded.

MR. DICKSON: It's not precluded.

VICE CHAIR PETTI: I think that's the better -- that's all --

MR. DICKSON: Yeah, okay.

VICE CHAIR PETTI: Anyhow, we're just trying to focus -- I mean, some of these designs, this is an issue that will require a different focus than if you were talking about larger reactors, because of the uniqueness.

MEMBER KIRCHNER: Is single failure going to be a requirement in --

MEMBER MARTIN: You have to have something in there, though, on single failure, like --

MR. DICKSON: Yeah.

MEMBER MARTIN: Like a definite "maybe"?

(Laughter.)

MEMBER KIRCHNER: Yes. I didn't have

enough time to read through that.

MR. DICKSON: Yeah, the regulatory history of single failure is -- it goes a long way back.

MEMBER KIRCHNER: Yes.

MR. DICKSON: And so, there's a -- I'll go back to an SRM-SECY paper. It was developed back in the seventies, 77-141 or 144; I can't recall.

But it is where the Commission, effectively, told the staff how to utilize single failure methodologies, and they were pretty clear that single failure was never really intended to be a method where every single component has to be single-failure-proof. But rather, it's a method to assess a system.

And that is what is going to be carried forward in Part 57, is that it's a method to assess a system rather than what some may think is -- that every component has to be single-failure-proof.

MEMBER KIRCHNER: It becomes -- first of all, it was a surrogate four or five decades ago for --

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: -- what we would now think of as lines of defense.

MR. DICKSON: Yeah. Yeah.

MEMBER KIRCHNER: You just presume that an important component --

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: -- is unavailable or it fails in the wrong direction, or whatever, and then, do a bounding design basis --

MR. DICKSON: Uh-hum.

MEMBER KIRCHNER: -- accident analysis. So, it was a crude way to do lines of defense.

MR. DICKSON: Right.

MEMBER KIRCHNER: And arbitrary.

Were it to be applied here, not to belabor the reactivity accident category, but it's worth belaboring here because a single stuck rod has a lot of worth. You may not get subcritical with the remaining drums or control rods, if one is stuck, on demand. That's even short of ATWS. So, that would be design-dependent.

VICE CHAIR PETTI: At least in the designs that I've seen, that's not an issue. It could be, I suppose --

MEMBER KIRCHNER: It could be.

VICE CHAIR PETTI: -- on the design.

I thought, when I read through the single-failure criteria of this and what they're

doing, it's leaning more towards the way LMP looks at things in terms of looking at a number of potential options, not trying to be prescriptive on which single failure you need, but to go through a thought process to figure out what is a credible failure.

MEMBER MARTIN: And safety-related components are still diverse and redundant.

MR. DICKSON: That's right.

MEMBER MARTIN: Single failure is maybe more narrow definition or, really, a more specific kind of diversion-redundant situation.

But LMP, of course, allowed that you could have, you know, well, two diverse -- I mean, two systems, diverse and also redundant, that do the same thing, but look completely different. And you can lose one, and you still have the other. And that covers you for the safety-related requirements on the system; you get that backup.

The looser that we, historically, dealt with LWRs.

MR. DICKSON: Uh-hum.

VICE CHAIR PETTI: So, there is a good discussion in the implementation plan about, okay, you've got three trains, and does the single failure have to take out the third train? They go through

some good sort of practical examples.

One, cooling system space, I mean, you have the protection system space. And there's at least stuff there that I think they can look at and think about what's the underlying rationale there, as they think about it.

MEMBER MARTIN: I'll give you an example of what was done for Pele. My approach to the safety case was, not initially, but it became risk-informed, right? So, we developed a PRA. We identified, basically, credible events, a set of credible events --

MR. DICKSON: Uh-hum.

MEMBER MARTIN: -- and the LBEs associated. Defined them as they would otherwise be defined from the PRA. We added single failure and we looked at a hundred of those. Right?

So, we applied AST on the dose consequences. Now, we would evaluate what was damaged. It's a little bit different for a gas reactor, a little bit, maybe in some ways, easier to do the problem, rather than an LWR.

But, nonetheless, for the hundred cases, one of them just naturally becomes the MHA --

MR. DICKSON: Uh-hum.

MEMBER MARTIN: -- because I basically have all the characteristics of an MHA. I don't know a priori going in what that MHA is, but I do have a PRA that gives me a good idea. I shotgun a hundred cases to determine the success. I put a single failure in there to kind grab that. And then, boom. And I was still, like, an order of magnitude or two below our acceptance criteria.

MR. DICKSON: Uh-hum.

MEMBER MARTIN: And I have to believe that anything that really qualifies as a microreactor would look a lot like what the Pele risk profile looks like.

MR. DICKSON: Uh-hum.

MEMBER MARTIN: And I see what you're allowing is more move to the left on that frequency consequence curve, which moves you farther and farther away from -- and it goes, in my mind, it could go far enough away, that you would almost argue, "Why do I need the NRC, you know, at some point. I'll just go to the EPA." You know?

So, I've already kind of ranted before, but I do feel like some factor of safety needs to be -- some guardrail needs to be incorporated. You know, it doesn't have to be an MHA, but some acknowledgment that you need to capture that epistemic uncertainty.

It may be in the way that ASME does with older pressure vessels. In 2026, they still -- we know a lot about what happens when things break. And yet, we still have the (audio malfunction). And that's, more or less, negotiated by a committee like this and the industries they work in.

CHAIR HALNON: Since we returned to where we began, can we move on?

MEMBER MARTIN: Sure.

MR. DICKSON: So, I'll take that down as assessing some other type of safety factor or margin that needs to be maybe more explicitly stated in the guidance document. It is, in effect, I think --

MEMBER MARTIN: It's a soundbite.

MR. DICKSON: This is what I'm getting, yeah. All right.

Okay. So, slide 34.

We're going to shift the focus on how to establish the site boundary using either the MHA or MCA analysis. This is a key step, because, under Part 57, the 1-rem eligibility criteria does apply at the nuclear plant level. So, that can be one reactor or it can be several reactors within the nuclear plant site boundary.

So, the first point here is that the site

boundary determination does become more complex as there's more units increased at the facility, as the physical layout becomes more integrated. A single-unit configuration would be rather straightforward, but a multi-unit, large area deployment introduces questions about combined source terms and shared SSC behaviors.

So, Part 57 does allow reactors to have either their own MHA or MCA, unless there's shared safety systems which can credibly impact one or another. If there are these shared systems, then they need to be assessed in the systematic risk evaluation process described in the regulatory guidance document.

So, we can, then, go on to the first example. So, what I've done here on slide 35 through 38 is provide a couple of visual examples. Pictures are worth a thousand words.

So, slide 35 illustrates the simplest case. It's an established site boundary under Part 57. It's a single-unit configuration. This is, typically, the starting point for understanding how the accident analysis translate to site layout.

For a single-unit plant, the radiological analysis is straightforward. The applicant defines the site boundary with a simple radial distance from

the reactor center or the most conservative release point or the reactor itself.

So, this is shown in figure 12-4 to help visualize this. This is a figure directly coming out from the regulatory guidance -- oh, sorry -- the NUREG Implementation Guidance. It conveys that the concept for a single reactor and a site boundary is, essentially, a circle. So that, even under the conservative assumptions of the MHA or MCA -- which there's questions to that, which one is the best route to go -- the site boundary dose still remains under the 1-rem eligibility criteria.

And this figure is just really the baseline for more complex arrangements. So, through the next slides, we'll talk about or show more complex arrangements.

Let's go on to slide 36.

So, slide 36 introduces a more complex configuration. It's a multi-reactor design that's deployed over a large area. So, each reactor has its own independent safety systems and they're not shared. In this deployment model, you can think of this as being maybe more of a fleet-style or energy hub type of fleet deployment business case.

In this arrangement, each reactor is

treated as its own independent radiological source. That means that the units can its own site boundary using their own MHA or MCA to demonstrate their compliance with the 1-rem acceptance criteria.

The slide notes that spatial separation is important in this configuration. So, each reactor has to be sited far enough from, you know, unit A versus unit B. In this case, it would preserve the integrity of all of the independent safety systems and the operational programs for each of those reactors.

Any questions on this one?

VICE CHAIR PETTI: So, as I recall, there's nothing in the implementation about what that distance should be.

MR. DICKSON: Between reactors?

VICE CHAIR PETTI: Between those, yeah. I mean, basically, they look as three isolated --

MR. DICKSON: Yeah.

VICE CHAIR PETTI: And, in general, I'd say, yes, except external hazards where one could get concurrent releases.

MR. DICKSON: Well, I mean, these could be -- well, the safety case is enclosed within the site boundary.

VICE CHAIR PETTI: So, they'll have to

consider all of them?

MR. DICKSON: There's another example where --

VICE CHAIR PETTI: That's right; that's what I remember.

MR. DICKSON: We're going to get to the next example where they have maybe a shared safety system, and that you have to come up with a plant-level MHA or MCA that would consider the impacts to each of them.

But the one I think that you might be referring to would be the fourth example. Okay, so slide 37.

This introduces the next level of complexity, multi-unit nuclear power plant reactor shared safety system. So, in this case, there's just a simple box between three different reactors. You can think of that, maybe that's the control room perhaps.

This scenario fundamentally changes how applicants would establish their site boundary, because a single failure could impact several reactors. And as we have it laid out in the Implementation Guidance, this is actually one of the categories of design basis accidents that you would be

assessing, this shared-system type of analysis.

Okay. So, with that, I can go on to slide 38.

So, slide 38 is probably the most complicated. This is the multi-unit plant where all of the reactors within the site boundary meet 1 rem. So, slide 38 extends the previous slide's discussion. This would probably be the most complicated approach.

You need to look at the impact to all the different reactors. It could be a seismic event, for instance, that would cause all of these separate units to have a maximum credible accident.

But that doesn't mean that we're -- and we're not alluding here that you would have to apply an MHA to every single one of these reactors. Right? You've got to find what impacts them all. You know, not this, like, volatile source term that goes up into the air and use that to site. But again, there is an analysis process that we've described in the guidance document to help cover this type of an arrangement.

Uh-hum?

DR. BIER: Yeah, coming back to my earlier question, would the licensing treatment of this arrangement be different if the multiple plants had different owners versus the same owner?

MR. DICKSON: I don't think so. It would all be the same, right, in our accident -- no, I do not think so. The treatment would be the same.

MEMBER HARRINGTON: This is Craig.

So, for the treatment to be the same, owner X would at least bear some -- not responsibility; that's not the right word -- but have to deal with the fact that Y and Z are right over there?

MR. DICKSON: That's right. These would be business cases that they are making. I mean, it would make more sense, to me personally, that you have a microreactor that has a small site boundary, and you can place a lot of them in one location and not have to share one large area.

Some of these designs, at least I've heard, they're saying that they have site boundaries of 100 meters.

MEMBER HARRINGTON: I realize this figure is representative, but --

MR. DICKSON: Yeah.

MEMBER HARRINGTON: But that's really 100 meters -- it would be 100 meters, the closest reactor, right?

MR. DICKSON: Well, that's on a

per-reactor. So, like, often, what the designers are saying, "Our reactor has a 100-meter site boundary." In this case, you would have -- you know, one, two, three, four --

MEMBER HARRINGTON: It would be 100 meters from the nearest --

MR. DICKSON: Yeah.

MEMBER HARRINGTON: That one being the circle.

MR. DICKSON: Yeah. Well, yeah, this is just a visual.

MEMBER HARRINGTON: Yeah.

MR. DICKSON: I mean, I don't know if you could ever find somebody that will actually, like, sell a plot of a land that's a perfect circle.

MEMBER HARRINGTON: Well, you could (audio malfunction).

MR. DICKSON: Yeah, from each one, right.

And that's, effectively, the presentation for my section. Then, there's still --

VICE CHAIR PETTI: Yeah, maybe we should go back to --

MR. DICKSON: Deployment models, Duke's.

MR. KENNEDY: Okay. Thanks. This is Duke Kennedy of the NRC staff.

All right. So now that we've figured out how you get into the rule, I'm going to talk about what happens, what options are available, once to be licensed under Part 57.

So, we only included the licensing instruments that we thought were necessary to enable the microreactor deployment models that we've heard of or that we could think up on our own. This is how we decided that we would allow deployment under a Construction Permit and an associated Operating License; that we did not need to have a combined license. This CP and OL process we thought could be executed faster than the COL process and we didn't need to provide flexibility in that respect.

We included a Manufacturing License, so that companies that didn't want to actually deploy reactors could get their designs approved and they could manufacture those reactors, and that could support another business case.

And we also included the Standard Design Approval, so that entities that didn't want to actually build reactors or manufacture them could still get a design approved to a certain extent.

As I mentioned before, we didn't see a need to include a design certification because the

generic finality that's available under a Construction Permit and Operating License would, essentially, provide the same level of finality as a design certification, as would a Manufacturing License that specified an entire plant design.

Also, again, we didn't include a Limited Work Authorization or Early Site Permit because we didn't see those as meaningfully speeding up the overall deployment timeframe for these smaller, simpler reactors.

Part 57 also allows for combining requests for various licenses and permits. So, an applicant that really understands and has their whole deployment model ready to go can submit a single application that would cover, say, a Manufacturing License, a Construction Permit, and an Operating License, Transportation Package Certification, Part 70n License for fuel, and so on.

So, the thought here is that we can reduce some of the administrative burden by allowing explicitly in this rule for all these different types of applications to be combined into one. I think this is especially important where you have transportable reactors and the reactor itself is functioning somehow as part of the transportation package. Wouldn't have

to repeat a lot of the same information in the Transportation Package Certification application, as well as, say, the Manufacturing License application. There's some efficiency there.

Next slide, please.

CHAIR HALNON: Wait. Just for a clarification -- and I think Elijah said it -- he said the SDA was a staff-level approval. What does that mean?

MR. KENNEDY: So, this is the same as it is under Part 52 and Part 53. So, Standard Design Approval is reviewed by the staff. It's reviewed by the ACRS, and then, there's a Safety Evaluation review, but it's not a license.

CHAIR HALNON: So, it doesn't go to the Commission?

MR. KENNEDY: Yeah, and it doesn't have a hearing.

CHAIR HALNON: Okay. I got you. Thanks.

MR. KENNEDY: Okay.

MEMBER PALMTAG: Would a Manufacturing License have to be tied to a CP or an SDA? Or could you have something where you put bounding, reactor type, for the Manufacturing License?

MR. KENNEDY: So, a Manufacturing License

can stand on its own. It requires the application to include the complete design of the manufactured reactor, as well as some information about the manufacturing process.

MEMBER PALMTAG: Do you happen to know that information offhand? So, you would need a CP to get the Manufacturing License?

MR. KENNEDY: No, the Manufacturing License itself is an approval, a review and approval of the design.

MEMBER PALMTAG: Okay. So, you'll need a separate -- you couldn't have a boundary, you know --

MEMBER HARRINGTON: It just produces a reactor out there in a box on the dock ready to go. Where you go with it is a different question.

MR. KENNEDY: So, they're all separate.

MEMBER HARRINGTON: Explain how would a CP be different. How could you get a CP without having it?

MR. KENNEDY: You specify the design of the reactor in the Construction Permit and Operating License application. So, if you look at Subpart C --

MEMBER PALMTAG: So, it would be, basically, (audio malfunction) here? I'm just trying to understand all the different options here.

MR. KENNEDY: Yeah, so the CP and the OL require the application -- the joint application requires the most information, all the information that we need in order to make the findings that the reactor can be constructed and that it can be operated.

The ML scales that back a little bit. You don't have site-specific information, but you have all the design information for the reactor needs to be provided in the Manufacturing License application.

The SDA scales that back a little bit further. Now, it's just maybe a major portion of the design.

So, if you go back to the joint application for the Construction Permit and Operating License, in that, you can reference an ML. So, any design review that's been -- the design review that's been done under the ML does not get repeated in the Construction Permit and Operating License review. All that design information that would normally need to be directly in the CP has now already been reviewed in the ML. So, by referencing the ML, you don't have to include it again.

MEMBER KIRCHNER: But is there equivalency to referencing an SDA?

MR. KENNEDY: So, for referencing an SDA, it's not exactly equivalent to referencing an ML, because, again, the SDA is a staff review. You don't have the same regulatory finality that you get from a Manufacturing License, because a Manufacturing License is subject to a hearing.

MEMBER KIRCHNER: You kind of widened the goalposts here with the entry criteria. So, it's very likely that we are going to see people try and test you and the entry criteria with much larger reactors that are (audio malfunction) and not manufactured.

So, could they have a -- could they do a direct CP at that point?

MR. KENNEDY: Yes. Uh-hum, yes. The reactors that are field-constructed fit under Part 57. There is no requirement for a Manufacturing License.

MEMBER KIRCHNER: Okay.

MR. KENNEDY: There's no requirement for an SDA. So, somebody could submit a joint application that is for a reactor design, and it could be a small one; it could be a medium-sized one, whatever. And then, they can go ahead and use that as the single licensing mechanism to construct the reactor, and then, to put the reactor into operation. There's no

requirement for anything to be manufactured anywhere else.

MEMBER KIRCHNER: Okay.

MR. ARAGUAS: This is Christian Araguas with the NRC.

So, just to go back to your question, Scott, I think what you're seeing here is the optionality in the rule. Right? If you want to get to fleetwide deployment quickly, you're going down through, potentially, the Manufacturing License path. But if you're not, if you're an entity that just wants to build a few reactors on a site, you don't necessarily need the Manufacturing License. You just go -- you know, your joint application gets your Construction Permit, your Operating Licenses, and you go. So, it's --

MEMBER PALMTAG: (Audio malfunction) --

MR. ARAGUAS: Yeah.

MEMBER PALMTAG: -- possibly CP, SDA, OL?

MR. ARAGUAS: Yeah.

MEMBER KIRCHNER: But there are concepts -- this is Walt Kirchner -- you have concepts out there for fleet deployment that likely may be field-constructed there --

MR. ARAGUAS: Yeah.

MEMBER KIRCHNER: -- being cooled SFRs that aren't transportable very readily and manufactured intact, and shipped as such. Okay.

MEMBER KAMMERER: So, some of the manufactured reactor designs, it doesn't come out as an entire reactor that's built, right? It's piece parts that are all manufactured, and then, assembled on the location. So, that assembly -- and in a lot of cases those are fueled onsite as well -- so, the assembly and ensuring the quality of the assembly, and then, ultimately, fueling it, is that all contained within the CP, that assembly-plus-shield. They almost all need shield structures. Well, they all need shield structures additionally. So, it's all of the rest of those parts? It's everything except that nuclear, that fabricated nuclear system itself, as in the CP and the OL?

MR. KENNEDY: That's correct.

MEMBER PALMTAG: The only difference on an ML would be you're actually putting it, all the pieces together at the factory? But if you're putting the pieces together at the site, that would be a CP.

MEMBER KAMMERER: The whole part of it is still made in a factory. It just has to -- it has assembled onsite.

MEMBER PALMTAG: Correct. So, that's the difference between the ML and the CP? It's just where -- how much gets assembled?

MR. KENNEDY: No. The CP has a specific site and requires site characteristic information. Because it's more --

MEMBER PALMTAG: But if you could bring all the pieces together and you assembled it onsite, that's a CP?

MR. KENNEDY: That's right.

MEMBER PALMTAG: But if you assemble it beforehand in a factory, then that would be an ML?

MEMBER KAMMERER: You still have to do site construction, no matter what.

MEMBER PALMTAG: Not necessarily. You could ship it through the (audio malfunction) and assemble it, and ship it to the --

MEMBER KAMMERER: But you would still have to put it on a pad. You still have to put a shield building around it. There's always construction. There's always going to be some construction that's tied to the site.

VICE CHAIR PETTI: Yeah, even though it may be minimal.

MEMBER KAMMERER: It might be minimal, but

you're still -- there's always going to be something.

MEMBER HARRINGTON: But an ML could cover four pieces that would still have to be assembled, but they could still be built under an ML?

VICE CHAIR PETTI: But, no, again, the ML hasn't anything to do with nuclear material. If they do everything else and they don't do fuel at their ML, that's a different story. Then, it looks like the guy who provides pressure vessels to LWRs, right? They're just in a machine shop, you know, a high level that is producing all this stuff. There may be some assemblies. It's when you bring the nuclear material on that you need --

MEMBER KAMMERER: Except I think that the ML provides an evaluation of the entire design for safety, including, ultimately, the fuel that will be in it.

MEMBER HARRINGTON: Yes, let's say they do the pellets. We're just speculating.

MR. KENNEDY: No, I'm sorry to say there's been a number of inaccurate statements.

The Manufacturing License is a license to possess a utilization facility. And it's an approval of a design of a utilization facility. So, if you're talking about just major components, that's not a

Manufacturing License. Okay? That's something else.

Major components can all be assembled together into a utilization facility at a manufacturing facility, covered by a Manufacturing License or at a site that has a Construction Permit.

The thing that's made in a manufacturing facility is a utilization facility. So, then, that can then be taken to the site of a Construction Permit to be completed with whatever other systems need to be built to support its operation, and then, be authorized to operate at that specific site.

A Manufacturing License has no site-related information. So, there might be an address where the manufacturing facility is going to be located, but we don't do any kind of a review related to that manufacturing facility. It's strictly the design of the manufactured reactor, some QA requirements, and other organizational requirements for the manufacturer of the reactor, and a few other limitations on what can be done with the manufactured reactor.

MEMBER KIRCHNER: So, do you consider external hazards for the manufacturing facility?

MR. KENNEDY: No. So, if fuel were to be loaded in the reactor at a manufacturing facility,

that fuel would need to be covered under a Part 70 license.

MEMBER KIRCHNER: Right.

MR. KENNEDY: And the Part 70 license is where things like that would be considered. So, the manufactured reactor at the manufacturing facility is treated sort of like a storage location for the fuel under the Part 70 license. It's a very specific storage location, which, hopefully, the intent is with the features to prevent criticality; that that location is, more or less, like a storage cask in terms of its safety profile.

So, any analysis that needs to be done to ensure that that fuel is going to be safe at the manufacturing facility is done as part of the Part 70 license, not as part of the Manufacturing License.

If you want to test the reactor at the factory, you would need an Operating License.

MEMBER KIRCHNER: I see.

MR. KENNEDY: You can't do that under a Manufacturing License and Part 70 license. You need a full Reactor Operating License to do the testing, nuclear testing.

MEMBER KIRCHNER: Yeah.

MEMBER KAMMERER: So, I want to clarify

something, because what I was talking about is not components. It is you put them all together; you have an operating reactor, or everything, except the fuel. So, are we on the same page?

Because there are some, let's say, there's five palletized parts and it's got everything except the fuel, because it actually is doing fuel separately. It means that you can get more power out of them. A lot of times it's heavy; it's hard to, you know, transport. So, those five -- that is a functioning reactor once you put fuel in it. Those get an ML, can get an ML. If they want fuel, they have to get a separate license.

And I appreciate that you all have allowed you know, put in elements to allow them to fuel onsite. I think it's very helpful.

So, we're still talking about the same thing, so they have this reactor in pieces. Everyone is going to have to get a CP and an OL, or they can't turn it on, at whatever facility, ultimately, that that thing is going to sit at.

MR. KENNEDY: Correct.

CHAIR HALNON: Is this any different than Part 52 now?

MR. KENNEDY: The Manufacturing License

provisions? Not really.

CHAIR HALNON: Okay. Can we move on?

I mean, we're just at a deficit for understanding how the world works. This doesn't have anything to do with Part 57.

MEMBER MARTIN: Well, you can't load fuel in Part 52.

CHAIR HALNON: Right. That's the only difference.

MR. KENNEDY: Like I said, the Manufacturing License under Part 57 is almost the same as under Part 53, just a few tweaks for these smaller technologies.

CHAIR HALNON: I just feel like we're spending too much time on trying to understand some nuances that don't really matter.

MR. KENNEDY: Okay. So, as I've mentioned, there are expanded provisions for finality. So, finality can be granted through a Manufacturing License or through a Construction Permit and Operating License.

Part 57 also increases the flexibility for making changes. Currently, you need to have an amendment to make any change to the design in a Manufacturing License. Part 57 would allow for

50.59-like changes to the design of a manufactured reactor. It also streamlines some other amendment processes, and it takes away the requirement for a special justification in the case of a departure from an already approved design.

So, if somebody wants to deploy a manufactured reactor, but they want to change some small part of the design, they have to show that that design is still safe with that small change, but they don't have to provide this additional special circumstances justification.

Go to the next slide, please.

So, there's options for site approvals. There's one site for one reactor. You can have multiple sites specified for a single reactor, so that reactor could be moved between sites over its lifetime. You can have multiple sites for multiple reactors or you could have a large geographical area for one or more reactors. So, these are all different approaches to siting that can be specified in a single Construction Permit application.

So, to get back to the earlier question regarding large geographical areas and disaster response, so the thought here is that it's possible that a state could decide that it wants to have

microreactors on hand to deal with emergencies on its own. So, it doesn't necessarily involve federal response in that sense.

And in terms of how big these large geographical areas could be, the rule as it is specified, it has to be within NRC's jurisdiction, so, potentially, the entire United States. But as the areas get bigger, it requires more and more information to be provided upfront. It expands the audience for the hearing. It expands the number of state and tribal interactions that need to happen. So, it's really up to the company to decide what does it really need to do and is it worth the effort.

DR. BIER: Vicki Bier again.

A quick clarification. You had mentioned that, in the old system, somebody would have to give a justification for wanting to make a design change. Was that burdensome? You know, it is more than just three sentences saying, "We wanted to operate at a different range of power levels," or whatever the reason is?

MR. KENNEDY: I am not aware of any examples related to that. But these criteria exist under Part 52 right now, and it's not -- to me, it did not seem as simple as saying, "We have a better way of

doing things for our business case. So, we want to do it that way." It was it needs to be an improvement in safety or some other things.

DR. BIER: Okay.

MR. KENNEDY: But we felt, with these smaller, simpler designs, that it would be appropriate to have more flexibility for design changes that might be needed, or as new models of the same reactor came out, we wouldn't necessarily need to review everything from scratch or require these special justifications, when they could make an adequate safety demonstration.

DR. BIER: Okay. And I guess knowing the reason may also not be important for assessing whether it's safe or not to do it. Okay. Thank you.

MR. KENNEDY: Thank you.

MEMBER KIRCHNER: And should they want a power uprate, which almost every vendor, sooner or later, always wants to do, you would have to still meet the entry criteria?

MR. KENNEDY: Yeah.

Okay. Then, next, I'll just talk about the General License for construction. So, in the rule, it establishes a General License and it sets conditions for when somebody becomes a general licensee. So, a General License isn't granted to

individuals. It's just something that exists that they can qualify for.

The first criteria is they have to have a docketed application, and that application needs to be for such a nth-of-a-kind reactor, not a first-of-a-kind reactor.

Also, to be a general licensee, you have to have completed all of the state permitting processes and other consultations, and then, there are some restrictions on what can be done under the General License.

So, the General License allows construction of safety-related equipment. It allows construction of SSCs that are going to relied on to comply with security requirements. So, it really allows anything to be built at the site.

But it does not allow a utilization facility to be completed at the site, because you have to have an NRC license to possess a utilization facility. So, construction can begin -- construction on pretty much anything can begin, but it can't be finished entirely before the Construction Permit is issued.

And so, under Part 57, we have a requirement that the General License can only be used

when a manufactured reactor is referenced in the application. Because that manufactured reactor is essential to having a complete utilization facility, we put on a restriction that that manufactured reactor can't be brought to the site until the Construction Permit is issued. So, it just won't be possible for someone to complete utilization facility, but will be possible for them to build everything necessary to start operation once the manufactured reactor comes to the site and is connected.

So, this, again, allows people to get ahead on construction while they're waiting for the NRC to finish its review, but it's predicated on this being an nth-of-a-kind reactor. So, it's something that's already been successfully constructed and put into operation, and the NRC has already reviewed the complete design. So, there's a high likelihood that there wouldn't be any issues that would be identified during the review that would modify what needs to be constructed.

However, work under the General License is at the risk of the general licensee. So, just because you've started constructing under a General License doesn't mean that the NRC is going to eventually issue a Construction Permit if it finds an issue.

Next slide, please.

So, I think, just at a high level, this shows the process. And I think the biggest takeaway here is that everything done for issuance of the Construction Permit is all those boxes in the blue area, and then, everything in the green is actually constructing the reactor and issuing the Operating License.

And so, what this just kind of generally shows is that there is a really front-loaded process. You go through a lot of the -- you go through the hearings. You go through the ACRS review, an environmental review. You provide all the information upfront. And then, when it comes time to actually construct the facilities and put them into operation, there are many less steps involved.

You can go to the next slide, please.

And so, the main feature here to take away from this is, again, the blue box up top is everything that's reviewed in the Construction Permit phase of the review, and then, down in the yellow box, we're showing a Construction Permit that authorized at least four reactors there, authorized construction of four reactors. And in this case, there is some shared systems, and that's represented in the box on the

right in the yellow down there.

And so, those shared systems are tied to each of those individual reactors in the license. And this is important because, in the model where reactors are operated and then replaced, you don't want to have to deal with any kind of decommissioning or concerns about those shared systems in terms of them being covered appropriately by a license. So, they're, basically, tied to all the licenses at the site, and when a reactor no longer requires -- is no longer going to be operated, then, essentially, it can be disconnected from the rest of those shared systems and technical specifications in licensing space, so that it can be decommissioned as that single reactor, and to leave all the other support systems that continue to be needed at that site intact.

Go to the next slide, please. Just that one.

So, this is showing the example of where a Construction Permit covers a large geographical area instead of a specific site. And in the upper left there, you can see the joint application includes some additional information requirements.

So, there has to be site characterization and environmental information for the entire

designated area. So again, this goes back to the question of how large an area do you want to designate, because you have to provide information for that that's at least bounding for that entire area.

It also requires the decommissioning and fuel storage plans be submitted with the application upfront. That's optional for, say, a single reactor or a single plant at a specific site.

It also must include a site selection procedure. So, once the NRC has approved this large geographic area and the Construction Permit, we need to understand what that Construction Permit holder is going to do to verify that the specific locations it decides to build reactors at still meet all the criteria in the Construction Permit.

And then, it has to have a procedure related to the deployment of those reactors, so that we have an understanding of what it's going to take at each specific location to construct that reactor and be ready to place it into operation. And again, this is to try to make sure that we don't have any surprises for this type of a deployment model. We want to know a little bit more information upfront to provide these additional flexibilities.

In the purple box in the bottom left,

we've shown deployment at single fixed locations. So, each location is selected. The reactor is constructed. The NRC makes a finding of readiness for operation, and then, the OL is issued. And that happens repeatedly as sites are selected.

In the bottom right, we've just shown a little graphic on mobile reactor deployment. And so, the reactor -- they can select a specific site. They could deploy a reactor there. Then, they could insert features to prevent criticality, and it could be transported to another site that they've selected -- again, following this procedure that's required in the joint application.

With that, I will take any questions on these licensing pathways, and I think that concludes the slides.

CHAIR HALNON: Anybody have a question?

MEMBER KIRCHNER: It's a small technicality, but did you say, for a single reactor at a single site, it did not have to have a decommissioning plan in advance? Did I get that subtlety wrong? I may have gotten it wrong.

MR. KENNEDY: No, that's not wrong. So, they don't have -- the regulations on decommissioning in Part 57, there still has to be some

decommissioning-related information provided upfront, no matter what, but under this model, we want to know more details about what is the whole decommissioning plan for the reactor, so we can understand that there's assurances upfront that they're actually going to be able to remove these reactors that were, essentially, granting them a permit to just construct in many different locations. So, we want to make sure we fully understand how all these reactors are going to get removed eventually.

MEMBER KIRCHNER: Thank you.

MR. ARAGUAS: Can you move us to slide 39?

MR. DICKSON: Thanks.

So, for the most part, now we've concluded the presentation, and this is just scheduling.

The rule is now out for public comment, a 45-day public comment period, which ends June 15th, 2026.

We do have a public meeting going to be scheduled within, I think, the next two weeks or so.

And then, the estimated final publication date in The Federal Register is December 1st, 2026.

MEMBER HARRINGTON: This is Craig.

Just as a general comment, trying to define a box, very consistently define a box of what

fits under this rule and what doesn't, obviously, is a daunting challenge, as we've worked our way around that six times this morning, or this afternoon -- today.

Now, I guess what I'm arriving at, whether it's what you intended or not, is that, in reality, as a practical matter, if somebody wants to build a reactor, they're going to have to come to you guys, and you're going to bring some process, adjudicate, help them figure out, and do that in a fairly consistent manner across all such applicants, whether they fit under 57, 53, 50-whatever.

And maybe it's less important -- I think we've been trying to define that box for you maybe too rigidly, and that the reality is more aspirational and you're going to have to figure it out with the applicants as they come in. Is that fair?

MR. ARAGUAS: This is Christian Araguas from the NRC.

So, I think that's an accurate characterization. And I think, you know, the challenges I think we'll have is somewhat what we hit on this morning, where somebody is going through and they want to try to demonstrate that 1 rem at site boundary. They go through their MHA or MCA. And we

start having discussions with them and saying, "You know what? That's not really looking like what we were thinking. You may want to try something different."

I think where it gets to be a little bit easier -- and we talked about -- is the intent of the rule, right? When you think about the number of applicants, how many are going to be ready, design-ready, right out of the gate with all of the information to launch under the rule?

So, I think that starts to really self-limit who's actually going to be using the rule. So, I think that the pool of applicants that we're going to have challenges with won't be that extensive, but I fully expect -- or we fully expect -- we're going to have to provide that level of clarity.

It's going to have to be done through various public engagements, whether it's workshops, certain pre-application. So, that's been discussed among the -- and I'll start using the terminology -- the Office of Advanced Reactor leadership team, making sure that we're laying out a clear path for people to understand, you know, when there's value for going for 50, 52, 53, or 57. And so, that's just going to have to be an ongoing education process through the various

means that we have available to us.

MEMBER HARRINGTON: Yeah, it puts a little bit more of the onus on you in the future rather than being terribly prescriptive in the rule itself.

MR. ARAGUAS: Okay.

MEMBER HARRINGTON: Thanks.

CHAIR HALNON: Any other questions from members?

(No response.)

Okay. At this point, I'm going to open for public comment.

And again, as you had mentioned earlier, we'll take them in the order that we received the request. Try to limit to several minutes. At this point, there's not that many -- I think we have one person who emailed. So, there's no time restriction at this point.

So, I'm going to call on Ed Lyman to be our first commenter.

And, Ed, don't feel too constrained. If it starts feeling like it's getting too long, I'll let you know that.

(Laughter.)

MR. LYMAN: Thanks. I'll be brief.

This is Edwin Lyman from the Union of

Concerned Scientists.

So, I'd like to say I think the members asked a lot of good questions today, and I didn't hear a lot of good answers from the staff. So, I really hope that you will continue to press them to nail these things down.

I think -- and this has come up many times, but it's worth repeating -- Member Petti's observation about the entry criterion, in particular, the challenges of demonstrating that 1 rem, that could be one of the most hotly contested or controversial or difficult parts of the application. So, the idea that that would all be taken care of in some pre-application space that could be mostly out of public view is inappropriate.

And so, I think too much of this rule, this proposed rule, is based on presumptions about the safety and low risk of these reactors that have not been demonstrated yet. So, I'm just concerned about this circular pattern.

What are you going to do if an applicant comes in, meets the entry criteria with whatever they have at the beginning, and then, later, data is developed that shows they don't meet the entry criteria anymore? What's going to happen then?

And then, with regard to the second criterion, limits on fuel mass, I don't think that the staff has made a persuasive argument there. There's very little coupling between that fuel mass, the ultimate power level the reactor could achieve, the source term. It seems there are too many uncoupled links there.

And I would just like to pull that one example in the real world, and that was the Kiwi-TNT test in 1965 at the Nevada Test Site, which was 188 kilograms, a heavy metal core that was HEU. It was 900 megawatts thermal.

It was deliberately exposed to a reactivity transient. The core was cold. There were no fission products in it. The reactor exploded. A large fraction of that core was vaporized. And even without any initial fission product inventory, the dose rate at around 100 meters, or the dose, total dose, just from the plume passage was somewhere like 1,000 rads at 100 meters, and 3 rads at 600 meters.

So, you can see how, if there was initial fission product inventory, it would be very difficult to meet these entry criteria, if you consider that type of event.

And so, that, therefore, brings in the

connection to sabotage. And even though I know that that's not the Committee's purview, it seems that the sabotage and safety are inextricably linked in this type of system, because of the potential for an operator to induce this kind of reactivity transient.

So again, I'm concerned about too many a priori assumptions that have not been sufficiently validated and the potential for significant hazards to public health and safety from this rule.

Thank you.

CHAIR HALNON: Thank you, Ed.

So now, if folks would please use Teams to raise your hand, I'll take them in the order they come up.

Adam Stein?

MR. STEIN: Hello. This is Adam Stein from the Breakthrough Institute. I appreciate the opportunity to provide a brief comment.

The discussion here has been very useful to us, as stakeholders from the public, to understand the rule more in-depth. I wish that this level of detail was possible in a general public meeting, instead of just through an ACRS meeting, where I could ask questions of the staff directly. However, I understand that this is to provide information to the

ACRS, as you deliberate your letter writing.

I would like the ACRS and the NRC to consider a little bit more clearly how the single versus multi-unit plant is defined. At the moment, in the rule, it seems that a multi-unit plant would still have to meet the 1 rem at the boundary, but each reactor would be limited separately to the 10 metric tons of material.

And if the 10 metric tons of material is being used as a proxy for residual heat and other characteristics, then it might be less important to the actual accident sequences than having shared SSCs with a multi-unit plant, and therefore, might not be the most important proxy to be a second entry criterion for a multi-unit facility.

So, I would like you to consider that a little bit more in-depth and how that fits into whether the 10-ton metric is an appropriate metric here or not, and whether that is clear enough in the rule.

I do believe that the ACRS Committee brought up a lot of interesting nuances in the rule, some of which I had not yet considered. So, I appreciate that.

I think that there was a lot of discussion

about how Part 57 fits relative to Part 15.52, which is somewhat understandable because that's where more experience is at the moment. But I believe the staff was trying to make Part 57 more of an extrapolation or outgrowth of Part 53.

And I'm not sure that's entirely clear in the rule. For example, in the rule, at least as I read it, you can reference decisions from other agencies in Part 50 and 52, but does not clearly say Part 53. So, I think that you might want to take a look at that as well.

Thanks for the opportunity.

CHAIR HALNON: Thank you, Adam.

Any other members of the public raise their hand on Teams?

PARTICIPANT: No.

CHAIR HALNON: Members of the public in the room?

(No response.)

Okay. I'm going to give another 10 seconds or so.

(Pause.)

Okay. That means we can close the public comment period.

Just as a comment to everybody, the

transcript for this meeting will be expedited. Probably within a week or so, we'll have those. So, those will be available. We don't normally do that, but this is a special case. So, we're going to expedite those transcripts.

Dave?

VICE CHAIR PETTI: I just wanted to thank the staff. It's quite clear that a lot of work went into the rule and the implementing guidance under an incredibly short time window. And you are to be commended for that the rule is in as good shape as it is, given those constraints. You did a good job. Thank you.

CHAIR HALNON: Thank you.

And I think everyone of us would echo that. And I've heard that outside the ACRS as well, a very nice job, what you have done. And we've got a lot of work ahead of us, but I think that it's a very good foundational platform.

MR. BURKHART: Okay. Did you want to any more additional deliberation?

VICE CHAIR PETTI: I think if we took a break and then we came back, we could do some deliberation, and then, I could read the letter.

CHAIR HALNON: We will do that.

It's 2:32. At this point, we will break until 2:50.

(Whereupon, at 2:32 p.m., the foregoing matter went off the record and went back on the record at 2:50 p.m.)

CHAIR HALNON: Okay. We're back in session. We continue to have a quorum. I just want to first open it up to members and consultants if there's any discussion that we need to have before Dave reads the letter in the record. Need further discussion?

VICE CHAIR PETTI: Let me tell you. I did -- oops. I did add three items based on the discussion I heard and some people who whispered in my ear. One which is we had been looking for it. The one rem number is also tied to the dose criteria for NPUFs. So to be complete, I put that in. It's not a big deal.

This whole inconsistency between the two types of reactor operators relative to the passive and inherently safe reactor, basically, it's an inconsistency. And I think it's more wordsmithing than anything else. If they want to have the two types, they just need to be clear on something.

The broader comment that Craig had raised

about this need for guidance document, you've now got Part 50, Part 52, Part 53, and Part 57. And for some of these new applicants, what's the right load? How do I decide?

They're working on that. I think I need to acknowledge they're working on that. I quickly wrote stuff here in the break. But to say we would support such a thing because we have before.

And then I've been really bold to argue the letter talks about the -- not a big fan of the metric heavy metal criteria. But upon thinking about it more deeply, a coupe members said it's very diverse, the entrance criteria. There's one rem and there's a whole story there.

And then there's this 10 metric ton heavy metal. If you just focused it on microreactors, you'd have a more focused rule with more targeted requirements as opposed to -- and we understand the need for flexibility. But sometimes flexibility goes too far. And that's probably our strongest recommendation and whether or not we will heed to that as a committee. It's in red in the letter because those are the subjects that I heard.

CHAIR HALNON: Doesn't that go a little bit upstream? I'm thinking about the low consequences

with this. What if you do have a reactor that has --

(Audio interference.)

VICE CHAIR PETTI: Isn't that Part 53?

MEMBER KAMMERER: Can I say I feel like the class of reactors that we saw a lot of discussion today, and I really feel for what the staff had to think about because they're challenging. They are new, transportable, could be moved site to site, could be manufactured. It's very different.

It has a whole different set of consideration and scenarios of anything we've seen before. And so I think opening the aperture too far and not targeting this very unique Class 4 which guidance does not exist, it's a lost opportunity to actually focus in on the problem of this gap that we have for this new class of reactors that we have coming down the pike. And as they have to then develop the follow-on guidance and various documents, they're going to have to think about that edge case that they're allowing with that 10 metric ton which is not a transportable, manufacturable reactor. To me, it seems like a real missed opportunity to create a much more targeted rule for this thing which currently -- for which there's a gap.

VICE CHAIR PETTI: I like your words. So

remember them when we get to the editing stage.

CHAIR HALNON: Well, it's curious to me that -- and I know some of these -- we don't know how much it cost that goes through 57 or 53. We know pretty much what the Part 50 and 52 is. Someone's business case may be based on the fact that I could get through those expeditious, time probably more than money, most expeditious and flexible rule for my reactor.

So I might want to push it towards that. Part 53 certainly is quite open. But just postulate. Part 57, you get the time frame.

MEMBER KAMMERER: But functionally, the content that they need to develop and demonstrations that they need to --

CHAIR HALNON: Same.

MEMBER KAMMERER: -- are the same, why would it be a big difference?

CHAIR HALNON: That's why time is very important throughout the project because you're paying three, four thousand people to sit at the site for an extra six months. It's huge.

MEMBER KAMMERER: Okay. But they're also having to come in with a complete design.

CHAIR HALNON: That's the point, yeah. So

I guess my thought process is we don't want to pick winners and losers from the reactors coming in. But we may be very much screening them out very quickly, especially if you get on that edge case you're talking about. It jumps over into can't get to the one rem or I can't get to the 10 because after my -- I'm just trying to think. Are we --

(Simultaneous speaking.)

VICE CHAIR PETTI: Everything they talked about is 1537. That's what they drew everything from. So you got to think of those reactors, those NPUFs, the NIST reactor down the street like the MIT reactor, TRIGA reactors. I get a sense of what that hazard profile is.

I look at micros, particularly TRISO-based micros and I go, yeah, I think with the proper design, you can probably get there. But then there's these others because the title allows other low consequence that are very large reactors in footprint, in physical dimension, in power that I just -- I would never consider those in that range. And so there's that backdrop as well.

I'd never consider an MHTGR to kind of be filed in the same category as NIST. I mean, those are the -- I always like to have a mental number line,

pictures of the reactors, right? The Gen 2 systems here and then Gen 3s and way down here a TRIGA reactor, where do they fit on the line if the line hazard?

CHAIR HALNON: I don't think it's a problem when it's clearly --

VICE CHAIR PETTI: Right.

(Simultaneous speaking.)

VICE CHAIR PETTI: But the question is you then have to -- you have to fuzz up things because it's so broad.

MEMBER KAMMERER: Yeah.

VICE CHAIR PETTI: You can't be as targeted.

MEMBER KAMMERER: When I say edge case, I mean that 10 MT, right? Because there's potentially things that could go in there right -- squeak in right underneath. That is a very, very different -- that's very different than a microreactor.

We'd never say that edge case for what you could possibly squeeze in here. It's going to be a microreactor or transportable, manufacturable, or even just a stick build but a microreactor and a TRISO-based HTGR. I just -- I feel like it would be a missed opportunity to target this a little bit. The

doors -- the apertures are so open in flexibility that I think it ultimately may end up being less useful to the designs that would really benefit from a microreactor.

CHAIR HALNON: I think that you did have a section in the letter talking about entry criteria. That's --

(Simultaneous speaking.)

VICE CHAIR PETTI: Yes, yes.

CHAIR HALNON: -- these comments and whatnot.

VICE CHAIR PETTI: My sense is as I read the implementation guidance, I just saw a lot of Part 53 in there. Now I know they didn't use it. But just the whole how you got to the licensing basis, there's a lot of parallels.

Again, different words per se. But you look at it through the risk lens and you down select. You bin. That's all Part 53. And the more that things get blurred across the two, I just think that just invites questions. And the poor staff is going to have to be the traffic cop. And I said, is that really --

MEMBER KAMMERER: Fair.

VICE CHAIR PETTI: -- fair. And again,

even schedules and that licensing -- the quick licensing is really the micros as far as I know.

MEMBER HARRINGTON: Yeah, if you got something more complicated than micro, the chances of going through all of the design, putting the entire application together before you do anything else are pretty small. Financial, whatever, you got to get a contract and start building something, work in parallel as it's a very simple device.

CHAIR HALNON: The wording of the rule is driving it towards that. The only time it's feasible is if it's extremely simple. Do you want to read the letter in?

VICE CHAIR PETTI: Yeah, I think if we read the letter in, I think that'll help focus our thoughts.

CHAIR HALNON: Okay. Sandra, can you bring the letter up?

VICE CHAIR PETTI: And please make it bigger.

MEMBER MARTIN: I noticed you styled this much different than other letters, not the traditional background discussion.

VICE CHAIR PETTI: Correct. Yeah, but, well -- and if you'll see other letters, when you get

into rules like this and you're talking about different topics, I just think it's easier to pick the topic, bring the background, what's our comments, next topic, because of the breadth. Okay, subject, proposed rulemaking on licensing requirements for microreactors and other reactors with comparable risk profiles, 10 CFR Part 57.

Dear Chairman Nieh, during its 735th meeting held for May 6th to 8th, 2026, the Advisory Committee on Reactor Safeguards discussed with the NRC staff the proposed rule on licensing requirements for microreactors and other reactors with comparable risk profiles, Title 10 of the Code of Federal Regulations, Part 57. Our Reactor Safety Standards Subcommittee also examined this matter during an engagement meeting with the staff on October 7th, 2025. We also benefitted from the reference documents.

Conclusions and recommendations, the rule has many innovative changes to the regulatory framework to align the safety and security standards with a microreactor's assumed low risk. The rule allows multiple licensing pathways, correctly identifies critical safety function design criteria attributes for microreactors, enables flexibility in a variety of operational programs, and provides new

risk informed transportation options. Two, the rule uses an attribute-based structure to determine whether a proposed reactor is eligible to be licensed under its streamlined framework and is consistent with the Commission's Policy Statement on the Regulation of Advanced Reactors and cohesively aligns the attributes with a policy statement and the guidance on developing the associated principle design criteria.

Three, the heavy metal loading entry criteria does not adequately capture a microreactor's hazard. A reactor thermal power limit that reflects the upper bound of a microreactor would be a better metric and would prevent higher power reactors with different risk profiles from inadvertently qualifying under 10 CFR Part 57. Four, allowing remote and/or autonomous operation is a significant change in the rule.

It is difficult to ensure reactor safety with remote/autonomous operation given the challenge of ensuring the data communication pathways that would be required for remote control remain reliable in light of the cybersecurity threats and the still evolving maturation of artificial intelligence driven autonomous control systems in other industries. Background, this rulemaking aims to create a risk

informed performance-based regulatory framework for microreactors matching safety and security standards to their low risk. Key goals include expediting licensing for facilities with low potential radiological consequences, enabling high volume standardized approvals for microreactors and modular reactors. The proposal also considers general licenses for reactors of their components.

Microreactors shared design and site features with non-power production or utilization facilities, NPUFs. The new regulatory approach draws from existing NPUF rules, Part 50, one step procedures, Part 52, and regulations for advanced reactors, Part 53, forming a new set of requirements under 10 CFR Part 57. Part 57 uses an attribute based structure to determine whether a proposed reactor is eligible to be licensed under a streamline framework that closely aligns with the Commission's policy statement under regulations of advanced reactors.

Licensing pathways, the proposed framework offers flexibility to applicants through multiple licensing pathways with streamlined requirements proportionate to the level of hazard. Available pathways include a joint construction permit and operating license, a manufacturing license, a standard

design approval, and a standardized CP/OL for operating reactors of identical design at multiple sites. A primary goal of the regulation is to issue an operating license within 6 to 12 months after an application is accepted.

To achieve this expedited timeline, applicants must submit the final design, a final safety analysis report, and fully developed operational programs at the time of application. The technical requirements for the FSAR were developed by integrating elements of NPUF licensing with corresponding elements of 10 CFR Part 52 and 53. Manufacturing license, manufacturing licenses have been broadened to permit fuel loading and low power testing in the reactor at the manufacturing facility.

The rule sets forth requirements addressing criticality prevention measures, protocols for handling special nuclear material, Part 70 license, and procedures for in factory inspections and non-nuclear testing to verify that the fuel configuration aligns with design specifications. Additionally, the manufacturing license application must specify interface requirements pertaining to the rest of the plant about a conceptual design of the remaining components and contain all relevant site

specific design information necessary for NRC review and evaluation. The duration of the license is 40 years.

The manufacturing facility is also allowed to provide radiative fuel storage under a Part 72 license, either in a cask in the reactor itself. A detailed decommissioning plan may be needed if the facility serves as a decommissioning facility of multiple microreactors. Unique, novel, and noteworthy aspects of the safety design approach, recognition of low risk.

The unique safety design approach articulated in the proposed rule explicitly assumes an inherently low risk associated with microreactors. To determine eligibility under this framework, two fundamental entry criteria are established. First, the reactor must demonstrate that in the event of a hypothesized accident, the radiological dose of the site boundary remains low, below one rem, total effective dose equivalent, TEDE.

Second, the reactor must contain no more than 10 metric tons of heavy metal which includes thorium, uranium, and plutonium. Inherent safety characteristics, microreactors can benefit from inherent safety features that further reduce risk.

These include a low radiological source term in the postulated accident scenarios, low power density, and the mutual compatibility of fuel, coolant, and moderator when present.

Such characteristics facilitate accident prevention through passive heat removal mechanisms and self-limiting reactivity feedback. These design features collectively contribute to a simplified and more robust safety case for microreactors. Critical safety design criteria attributes, the safety case for microreactors is structured around six critical design criteria attributes: reactivity control, heat removal, fission product retention, shielding, radioactive affluent control, and security by design.

From these attributes, the facilities principle design criteria must be derived. And the safety analysis could be based on a straightforward maximum hypothetical accident similar to the approach used for research and test reactors. Defense in depth remains a fundamental requirement.

However, tailored risk mitigation strategies and outcomes from safety analysis may be employed to demonstrate sufficient defense in depth and safety margins. These attributes are intended to guide designs for limiting excess reactivity to

protect fission product barriers and toward maintaining core sizes that enable passive heat removal without reliance on emergency power, thus reinforcing the intrinsic and passive safety principles underlying the Part 57 framework. The attribute-based structure cohesively aligns the attributes with the Commission policy statement on the regulation of advanced reactors and guidance on developing principle design criteria.

Unique, novel, and noteworthy aspects of operational programs, operational flexibility for microreactor programs is introduced across several key areas: establishment of an general license or construction, enhanced flexibility regarding weather data, external hazard characterization and site characteristics when compared to conventional Part 100 siting requirements, implementation of a quality assurance regime modeled after research reactors rather than power reactors, Appendix B, 10 CFR Part 50, adoptions of a wider range codes and standards to allow for greater flexibility, utilization of closeout inspections in place of ITAAC, alternate fitness for duty protocols controlled by the licensee an overseen by the NRC, new Subpart B of Part 26, provision for categorical exclusion from environmental assessments

where applicable, transition from continuous onsite resident inspectors to periodic inspection schedules, decommissioning procedures based on methodologies employed for NPUFs, application of 10 CFR 50.69 criteria to determine the necessity of license amendments for FSAR changes, use of generally licensed reactor operators -- we'll need to change that because I wrote that before these new words -- allowance for remote monitoring, remote operation, load following, and autonomous control of the reactor and the implementation of a consequence-based cybersecurity program.

Operational programs concerning radiation protection, technical specifications, limiting conditions for operation, fire protection, and electrical equipment qualification will remain consistent with those implemented at the commercial nuclear reactor facilities. Unique, novel, and noteworthy aspects of transportation, the rule addresses microreactor transport from the manufacturing facility to the operating site. The fueled reactor can be treated as the transportation package and must meet Part 71 requirements.

Ultimately, the fueled reactor can be treated as approved contents in a newer existing

transportation package. The rule would allow safety evaluations to use risk informed assessments instead of specific testing, for example, drop testing, currently required in Part 71. Features to prevent criticality must be part of the package. Post-transferred inspections of the site are required to check for damage and ensure interface requirements are met.

Implementation guidance, staff -- I think it wants to say have developed -- detailed implementation guidance for 10 CFR Part 57. And it's a NUREG. I think it's 2217. This document parallels the standard review plan for large LWRs in NUREG 1537 for NPUFs.

The breadth and level of detail are noteworthy. Providing guidance for both applicants and staff reviewers is an efficient approach that delivers relevant information to each audience without requiring separate documents. The document also brings together in one place a coherent set of definitions that are sometimes used imprecisely when discussing reactor safety attributes.

In addition, the accident analysis chapter is comprehensive. It clearly explains the objective of each element of the safety analysis, for example,

evaluating the facility's ability to meet entry criteria, addressing siting and environmental consideration, establishing the ultimate capability of SSCs, and providing the basis for technical specifications. The chapter further explains how to integrate qualitative and quantitative risk informed approaches to characterize plant hazards to use hazard analysis to establish a range of design basis events, classify those events by frequency of occurrence, identify the limiting event in each event family to define design basis accidents, establish a maximum credible accident within this framework, and distinguish between a maximum hypothetical accident and a maximum credible accident, terms that have historically been confused and discuss how either approach can be used to demonstrate compliance with the one rem entry criterion.

The section on licensing pathways should be helpful for the range of deployment scenarios currently under consideration by applicants. ACRS comments, microreactors involving a number of different technical considerations than traditional large reactors. The appendix here provides a discussion of these considerations.

We offer the following comments on the

draft rule. We agree with the staff that the hazards of microreactors should be characterized in terms of entrance criteria that include both a dose consequence criteria and an additional deterministic criterion to further assure the risk is bounded, consistent with the assumptions that underlie a Part 57 streamlined licensing approach. The one rem site boundary dose aligns well with NRC's definition of a research reactor and 10 CFR 50.2, with the public dose criterion, use for NPUFs in 10 CFR 50.160, and EPA's protective action guideline for evacuation.

We have 10 metric heavy metal loading criterion. One, it is too high and has the potential to broaden Part 57 applicability to reactors with a higher risk profile than microreactors. And that should be a two.

Metric ton heavy metal is not the proper metric because it does not account for the differences in enrichment in the various applications and therefore does not directly represent fissile material inventory. Because the isotopic inventories that dominate the radiological source term, for example, iodine-131 scale directly with reactor thermal power. A reactor thermal power criterion that represents the upper limit of a microreactor would be a better metric

than heavy metal loading.

While we do not suggest a specific value of power level, it should be low enough that modules of SMR currently under development such as NuScale, Hermes-2, Xe-100, and larger reactors evaluated in the past, for example, MHTGR, would be precluded from such a designation because of their different risk profile. The rule could apply to a wide range of reactor sizes that meet the diverse entry criteria, one rem and 10 megawatt -- 10 metric tons heavy metal. Because of this broad scope across both technologies and sizes, substantial guidance is needed to ensure consistent application.

Limiting the rule, the microreactors would narrow the scope and allow more targeted safety requirements. Criteria to distinguish between an MCA and incredible events are not well defined. As stated in the implementation guidance, power reactors require assessment of anticipated transience without scram whereby the normal reactor protection system is deterministically assumed to fail even though an RPS is designed to safety system redundancy and independence requirements. An ATWS event such as an unintended broad withdrawal could be particularly severe for a microreactor where the control rods are

holding down a significant amount of excess reactivity leading to releases that exceed one rem entry criterion.

Neither the draft rule nor the implementation guidance is clear on whether such an ATWS event must be deemed to be credible on selecting accident scenarios to consider for inclusion in the MCA. The distinction between two operator types, those who can meet the one rem entry criterion without operator actions, GLROs, and those who require operator action, SLROs, seems inconsistent with the rule's intent. Specifically, the rules anticipated target reactors with passive and inherent safety features that typically do not require operator intervention.

The draft rule allows for remote and/or autonomous operation. This is a significant change from existing requirements as some members of the committee noted in added comments regarding the Part 53 rule in ML22196A292. Reactor safety for remote operation would rely on maintaining the confidentiality, integrity, and availability of the bidirectional communication link so that the reactor can be placed in a safe state when warranted.

How to adequately meet this cybersecurity

challenge is unclear. The unpredictable nature of an AI based autonomous control system is a related concern. Guidance provided with the draft rule addresses these topics only at a high level, although the federal register notice includes a question asking whether additional requirements or guidance are warranted.

We are interested in the staff's basis for confidence that from both a cybersecurity and software assurance perspective, a microreactor with remote and/or autonomous controls will be acceptably safe. We would appreciate the opportunity to further evaluate the acceptability of remote and/or autonomous operation in light of both changes in the rule resulting from the FRN question and any proposed design that the staff has concluded can demonstrate an acceptable degree of safety. Flexible NRC licensing pathways are valuable and reflect the range of application types expected in the future.

A guidance document outlining the available pathways, Part 50, 52, 53, and 57, help applicants determine which option best fits their reactor design. Summary, the rule has many innovative changes to the regulatory framework to align safety and security standards with the microreactors assumed

low risk. The rule allows multiple licensing pathways, identifies proper critical safety function design criteria for microreactors, enables flexibility and a variety of operational programs, and provides new risk informed transportation options.

Part 57 uses an attribute-based structure to determine whether a proposed reactor is eligible to be licensed under its streamlined framework that is consistent with the Commission's policy statement on the regulation of advanced reactors and cohesively aligns the attributes with the policy statement and guidance on developing principle design criteria. The heavy metal loading entry criteria does not adequately capture a microreactor's hazard. A reactor thermal power limit that reflects the upper bound of a microreactor would be a better metric and would prevent high power reactors with different risk profiles from inadvertently qualifying under 10 CFR Part 57.

Allowing remote and autonomous operation is a significant change to the rule. It is difficult to ensure reactor safety with remote/autonomous operation given the challenging of assuring the data communication pathways that would be required for remote control remain reliable in light of the

cybersecurity threats and the still evolving maturation of AI driven autonomous control systems in other industries. Sincerely, Greg Helm. Now there's a whole appendix that I probably need to read in. So let me take a drink of water.

DR. BIER: If people are making minor comments before you do the appendix or you want to --

VICE CHAIR PETTI: Let's wait. I think let's get it all ready because then we can release the court reporter. Reactor design safety and operational deployment considerations, background and introduction.

Consistent with the Commission's policy statement on the regulation of advanced reactors, attributes to be considered for microreactor design and deployment should include highly reliable and less complex safety systems using inherent or passive means to affect shutdown and decay heat removal. Longer reactor time constants, simplified safety and protection systems, infinite emphasis on accident prevention, reliable and maintainable equipment and systems, and designs that incorporate defense in depth to maintain multiple barriers against radionuclide release and designs that integrate safety and security requirements to eliminate or reduce potential theft or

diversion of nuclear materials. That's a partial listing.

This appendix contains additional considerations that are unique to microreactor design and deployment compared to larger fixed nuclear power plant installations, control of reactivity. Microreactors pose a number of unique reactivity challenges because of their smaller size and the desired objective of longer core lifetimes. Compared to larger SMR and LWR designs, the small diameters of microreactors generally entail neutronically leaky systems.

While this enables a combination of in-core and ex-core control elements, it also means that for under moderated thermal neutron reactor designs, water ingress or immersion or a change in the ex-vessel -- ex-reactor vessel configuration could lead to position reactivity insertion and feedback. Likewise for leaky fast-neutron reactor designs, additional ex-reactor vessel reflective material could result in positive reactivity feedback or diminishment in leakage depended upon for minimizing reactivity insertion over power events. In general, the small size of the reactor, the greater the worth of single control elements.

Further, because most reactors use a cylindrical core configuration, a smaller reactor head diameter practically limits the number of control rod/drum drive mechanisms that can be used. Also, the rod/drums have proportionately much greater worth than in larger designs of similar technology. Therefore, uncontrolled rod withdrawal, rod ejection, or drum rotation is potentially a much more serious reactivity insertion event than in a large reactor like a large PWR.

In addition, microreactor designs for long lived reactor cores result in a large amount of excess reactivity that must be held down for the life of the plant. In both of these cases, there is increased reliance on highly reliable control and protection systems to prevent inadvertent reactivity events from causing extensive core damage. The small size also results in harsher neutron gamma radiation environments and possibly higher thermal loadings on control elements, control rod drive and drum mechanisms, challenging equipment qualification and lifetimes.

Motors, bearings, seals, magnetic latches, indoor electrical switches, springs, or valves that are sensitive to dose and temperature can reduce the

reliability of the reactivity control safety function. Our configurations of some designs can lead to issues of bow and sag, both the core internals and the control rod drive/drum systems that must be accounted for to assure successful reactivity control safety function. Control of heat, control of heat generation and decay heat removal from microreactors presents conflicting design tradeoffs.

The large surface to volume ratio of a microreactor should make passive decay heat removal easier even if shielding is incorporated in the reactor vessel or as ex-vessel structure. However, smaller cores at higher power density generally have much higher peaking factors because of the radial and neutron axial flux shapes, presenting life limiting burn up fluence and thermal mechanical limits for fuel internal structures, vessels, and possibly for power conversion components. Likewise, at higher operating temperatures, meeting the thermal mechanical requirements of the ASME boiler and pressure vessel code, Section III, Div. 5 can be quite challenging, for example, a gas-to-gas heat exchange for a Brayton cycle.

And accelerated thermal creep/corrosion issues also need to be addressed. Trace of

structures, coolants, and heat removal working fluids, and power conversion systems can introduce the potential for chemical and energetic reactions and produce combustible gases that need to be considered. Retention of radionuclides, for microreactors adopting a function containment approach to control and retention of radionuclides.

Integrity of fuel and cladding, reactor vessel and primary cooling boundaries is critical, even in low pressure systems to prevent radionuclide release due to air and water ingress or chemical reactions, for example, energetic reactions, combustible gas generation, liquid metal fires, or salt interactions with structures. Siting and external events, current large light water reactors are typically enclosed in large hardened steel concrete containment structures ***3:26:53 large exclusionary boundaries and emergency planning zones, and deploy extensive in-depth security barriers and features with trained and armed security forces. In contrast, proposed microreactor deployment concepts envision rather small structural footprints, small EABs and EPZs, if at all, and minimal operational personnel if not quasi-autonomous mode operation.

Therefore, the combination of external

hazards safeguards security and environmental concerns pose much more challenging design consideration constraints for microreactors. External events may dominate the risk for microreactors depending on the site selected. Enclosures comparable to that of flex equipment, housing, and storage at existing LWR sites may be needed to provide adequate protection against external hazards.

The greater neutron/gamma leakage of microreactors makes the shielding safety function very important to prevent neutron activation of adjacent structures and placement structures and/or ground or air around the reactor. SHINE and preferential streaming leakage along axially along heat pipes is a major shielding challenge. The result in radiation fields from the reactor and activation of the power conversion equipment and the working fluids, for example, sodium, can impact worker dose during operation and maintenance and even during shutdown making inspection and repairs difficult.

Inadvertent exposure of occupational work in the public needs to be precluded. Stringent liquid and gaseous radionuclide affluent controls are needed to prevent significant atmospheric releases and potential contamination, for instance, tritium

contamination of groundwater. Design choices of materials are important to minimize neutron activation of structures.

High winds or tornadoes, flooding, seismic forces could dislodge and move structures, impact the microreactor, and potentially damage radionuclide barriers. Microreactors, fast and thermal, are susceptible to increases in reactivity due to immersion, acting as a reflector by flooding or due to intense precipitation of fire prevention response. Under moderated thermal and epithermal systems with the presence of internal void space are susceptible to increases in reactivity from internal flooding.

Transportation and deployment, factory fabricated and fueled microreactors present special challenges in preventing criticality and performing test/inspections during deployment. Additional requirements that physically lock mechanisms on reactivity control elements as well as added neutron poisons/absorbers for shutdown margin during transport may be needed, for example, to address accidental water immersion. Factory nuclear testing will likely be limited to approach to criticality and result in only limited confirmation of nuclear design characteristics to prevent build up of a source term

that would complicate handling transport and installation at a deployment site and have dose consequences in the event of a transportation or handling accident.

Therefore, comprehensive testing of power is likely only possible in the field. Because of structural and environmental loads during transport and erection. Most factory testing will likely need to be repeated, for example, of leak tag integrity.

In particular, inspection and functional testing and calibration against acceptance criteria of reactivity control mechanisms and elements and associated reactor protection system sensors and inputs, post-transportation loads, and installation is essential. Refueling, recovery, and decommissioning present specific challenges in the field. The fission product inventory and radiation fields of a high burnup microreactor core after operation will approach those of one to several spent of the refuel elements depending on the power level and burnup.

These high hazards will require protective measures and shipping containers that meet 10 CFR 61 performance requirements for packaging and transport of spent cores and activated materials. I am done. Thank you.

CHAIR HALNON: Thank you, Dave. So at this point, Allegra, I believe that you are complete for this meeting and for the rest of the week. We appreciate your services. Let us know if there's anything we need to do for you to get the transcripts out quickly.

(Whereupon, the above-entitled matter went off the record at 3:31 p.m.)

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

ACRS Full Committee Briefing

Proposed 10 CFR Part 57 "Licensing Requirements for Microreactors and Other Reactors with Comparable Risk Profiles"

Elijah Dickson, Ph.D., Senior Reliability and Risk Analyst, NRR
Michael Balazik, Senior Project Manager, NRR
William Kennedy, Senior Project Manager, NRR

May 6, 2026

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

Agenda

- Purpose
- Overarching Principles
- Rule Structure
- § 57.25 “Applicability Criteria”
- Part 57 Licensing Pathways
- Implementation Guidance
- Schedule

Purpose

- Current licensing frameworks are not optimized for high-volume, rapid deployment of microreactors and other reactors with comparable risk profiles.
- Establish a streamlined regulatory framework, supported by multiple licensing pathways, that leverages design standardization to efficiently license microreactors and other reactors with comparable risk profiles.
- The licensing process would complete, as much as possible, the safety, security, and environmental reviews with public engagement at one time, by leveraging a joint application for a construction permit (CP) and the associated operating license(s) (OL).
- Site reviews would verify that the site characteristics are within the reactor design envelope where pre-existing generic data can be leveraged as much as possible.
- The reactor design envelope would need to meet two entry criteria to justify an alternative risk-informed and performance-based regulatory framework for various requirements and operational programs (e.g., safety, security, environmental).

Overarching Principles of Developing Part 57

Laws

~ Atomic Energy Act of 1954, as amended, section 103

~ Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy (ADVANCE) Act

~ Nuclear Energy Innovation and Modernization (NEIMA) Act

Executive Order 14300

~ Sec. 5(e) Establish a process for high-volume licensing of microreactors and modular reactors, including by allowing for standardized applications and approvals and by considering to what extent such reactors or components thereof should be regulated through general licenses.

Commission Policy Statement and Staff Requirements Memoranda

~ SRM-SECY-98-144, "White Paper on Risk-Informed and Performance-Based Regulations" (ML003753601)

~ Commission's Policy Statement on the Regulation of Advanced Reactors (73 FR 60612, October 14, 2008)

~ SRM-SECY-24-0008, "Micro-Reactor Licensing and Deployment Considerations: Fuel Loading and Operational Testing at a Factory" (ML25168A133)

~ SRM-SECY-25-0052, "Nth-of-a-Kind Microreactor Licensing and Deployment Considerations" (ML25317A650)

Rule Structure

Focused Part 50 technical requirements and Parts 50 and 52 licensing and approval frameworks.

- Subpart A – General Provisions
- **Subpart B – Eligibility**
- **Subpart C – Construction Permits and Operating Licenses**
- **Subpart D – Manufacturing Licenses**
- **Subpart E – Standard Design Approvals**
- Subpart F – Reporting of Defects and Noncompliance
- **Subpart G – Irradiated Fuel Storage, Decommissioning, and License Termination Requirements**
- Subpart H – Maintaining and Revising Licensing-Basis Information
- **Subpart I – Transportation Package Design Certification**
- Subpart J – Physical Security Requirements
- Subpart K – Categorial Exclusion
- Subpart L – Inspections
- Subpart M – Material Control and Accounting
- Subpart N – [Reserved]
- Subpart O – Enforcement
- **Subpart P – Operator Licensing and Human Factors**
- Subpart Q –Reporting and Other Administrative Requirements
- **Part 26, Subpart P – Fitness For Duty**

Part 57 Subpart B - Eligibility

- **Intent: Attract reactor designs with high thermal and shutdown margin for rapid, high-volume licensing.**
- Utilizes two “Entry Criterion” and six “Design Criteria Attributes,” which describe attributes of microreactors and other reactors with comparable risk profiles.
- Not defining microreactors and other reactors with comparable risk profiles by power level.
- Given the wide range of reactor types and their functional characteristics, this proposed rule would emphasize these “attributes.”
- Attribute-based approach describes the capability to prevent or mitigate accidents without active systems or operator intervention.
- Recognizes certain reactors with inherently safe and passive design features with favorable safety profiles may appropriately be designed with higher power levels than other reactor designs.

Part 57 Subpart B - Eligibility (cont.)

Accident Consequence Criterion

10 CFR 57.25(a)

- ≤ 1 rem TEDE in the unrestricted area following the onset of a broad range of design basis accidents for their duration.

Fuel Mass Limit

10 CFR 57.25(b)

- ≤ 10 metric tons of heavy metal per reactor.

Design Criteria Attributes

10 CFR 57.30(a)-(f)

- Reactivity control
- Heat removal
- Fission product retention
- Shielding
- Radioactive effluent control
- Security-by-design

Section 57.25(a) Dose-based Entry Criterion

- A 1 rem TEDE dose-based entry criterion, based on 10 CFR 50.34(a)(1)(i), under accident conditions would be used to inform the analysis of postulated accidents and the development of safety measures so that, in the unlikely event of an accident, the NRC has reasonable assurance that no acute radiation-related harm will result to any member of the public.

- An applicant would be required to demonstrate their reactor design meets the 1 rem TEDE dose-based entry criterion in proposed § 57.25(a) using either maximum hypothetical or maximum creditable accident methodologies.

- **Bases**: The Commission has found the use of an accident dose-based acceptance criterion to be adequate for siting and design purposes based on decades of extensive experience in the criterion's application and in recognition of the conservatism applied within the radiological consequence analyses.

- **Purpose**: Site characterization and mitigative safety-related SSC sizing.

Section 57.25(b) Fuel Mass Limit Entry Criterion

- The total inventory of thorium, uranium, and plutonium contained in the nuclear reactor or any individual nuclear reactor that is part of the nuclear plant must not exceed 10 metric tons.
- **Bases**: experience gained from previous reactor reviews.
- **Purpose**: Deterministically screens reactor designs without additional performance-based acceptance criterion or severe accident analysis to assess events beyond which SSCs could be challenged.
- **FRN Question**: NRC has proposed a question in this proposed rule, asking whether, in lieu of applying a deterministic material limit on the quantity of SNM, the NRC should apply an alternative performance-based acceptance criterion (e.g., adiabatic heat rate threshold, or severe accident analyses).

Section 57.30(a)-(f) Design Criteria Attributes

Bases: leveraging concepts from the Commission's Policy Statement on the Regulation of Advanced Reactors (73 FR 60612; October 14, 2008) and experience gained from previous reactor reviews.

The applicant for a license or design approval under this part must provide information that demonstrates that the nuclear reactor or nuclear plant design has design criteria attributes that satisfy the following:

(a) *Reactivity control*. The design must provide for the following:

(1) Control of the power level during normal operations;

(2) Rapid insertion of reactivity control devices to immediately shut down the reactor and maintain it in a safe shutdown state under accident conditions; and

(3) Net negative reactivity feedback as a result of increased reactor power.

(b) *Heat removal*. The design must provide for highly reliable passive decay heat removal to limit core coolant and fuel temperatures during accident conditions to within design limits to protect the fuel and, as appropriate, the reactor coolant and fission product boundaries.

(c) *Fission product retention*. The design must provide for the protection of engineered fission-product boundaries to limit the fission product release of radionuclides during normal and accident conditions.

…

Section 57.30(a)-(f) Design Criteria Attributes (cont.)

(d) *Shielding*. The design must provide the following:

(1) Adequate permanent and temporary shielding to comply with 10 CFR part 20 for the protection of workers and the public from direct radiation exposure from the reactor and radioactive sources during operation, shutdown, and transport, including during abnormal conditions; and

(2) Sufficient robustness and heat removal to prevent loss of shielding integrity during normal and accident conditions.

(e) *Radioactive effluents control.* The design must meet the requirements of part 20 of this chapter for control, monitoring, and release of radioactive materials to the environment.

(f) *Security by design.* Safety and security must be considered together in the design process such that, where possible, security issues are effectively resolved through design and engineered security features.

Subpart C - Construction Permits and Operating Licenses

- Front-loaded review:
  - Require final design information and operational programs in a joint application for a CP and associated OL(s).
  - One opportunity for contested hearing on the joint application.
  - One ACRS review of the joint application.

- § 57.60(a) Contents of applications; technical information – *Final safety analysis report*. An applicant must:
  - Include a comprehensive safety analysis covering site characteristics, reactor design and safety features, remote monitoring/autonomous operation, accident evaluations, and assessments of safety-related systems, structures, and components.
  - Describe the details of operational programs, security, emergency, quality assurance, fire protection, and organizational programs necessary for safe construction, operation, and decommissioning of the facility.
  - Provide supporting technical information, including design bases, codes and standards used, risk analyses, effluent and radiation controls, criticality safety, and applicant technical qualifications.

Subpart C - Construction Permits and Operating Licenses (Cont.)

- § 57.60(b) *Environmental information.* An applicant must:
  - Provide information justifying application of a categorical exclusion, or if a categorical exclusion is not applicable, an environmental report or applicant-prepared environmental assessment

- § 57.60(c) *Request for generic finality.* An applicant may:
  - Include in its joint application a request that the Commission afford generic finality, in accordance with § 57.142(e), to the construction permit, associated operating license(s), or both.

- § 57.60(d) *Large designated areas.* Provides the information that must be included in a joint application for a CP and associated OL(s) that designates large geographical areas for construction and operation.

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Subpart D - Manufacturing Licenses

- A manufacturing license (ML) provides authorization to manufacture and possess reactors and affords finality to the design for the purpose of referencing in a joint application for a CP and OL(s).
  - Requires final design information for the manufactured reactor and information related to the manufacturing processes, organization, controls, and inspections.
  - One opportunity for contested hearing on the application.
  - One ACRS review of the application.

- § 57.155 *Contents of applications; technical information in final safety analysis report.* An applicant must:
  - Include a final safety analysis report covering the design of the manufactured reactor and specifying the site parameters postulated for the design.
  - Propose technical specifications applicable to the manufactured reactor.
  - Provide interface requirements between the manufactured reactor and the remaining portions of the nuclear plant *or* include all non-site-specific information on the remaining portions of the nuclear plant that would be included in a joint application for a CP and associated OL(s).

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Subpart D - Manufacturing Licenses (Cont.)

- § 57.160 *Contents of applications; additional information.* An application:
  - Must include information related to the manufacturing processes, organization, controls, and inspection.
  - May include standard operational programs
  - May include information related to loading fuel and the required features to prevent criticality

- § 57.197(d) *Fuel loading.* A holder of a manufacturing license may load fuel into a manufactured reactor pursuant to a license issued under part 70 only if the manufactured reactor is configured before its fuel loading and during storage and transport with features to prevent criticality that are specified in the manufacturing license

Subpart G - Irradiated Fuel Storage, Decommissioning, and License Termination Requirements

- Relies on the regulations in 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste,” which would provide a general license for fuel storage to Part 57 OL holders (the same as for Part 50 OL holders and Part 52 combined license holders).
- Clarifies the need for a specific license under Part 72 for storage of irradiated fuel at a location other than the site of a power reactor (e.g., a manufacturing facility).
- Allows for in-reactor storage of irradiated fuel, subject to the conditions of a Part 57 OL or a Part 70 license that would ensure safety of the fuel during storage.
- The provisions in proposed Part 57 related to decommissioning are generally consistent with the framework in § 50.82(b) for non-power production or utilization facility licensees.
- Applicants can submit decommissioning plans with the initial joint application for a CP and associated OL(s).
- Provides flexibility for the variety of decommissioning strategies that may be included in deployment models, including site-specific decommissioning cost estimates, dismantling the reactor at a location other than the deployment site, and refurbishment and redeployment of reactors.

Subpart I – Transportation Package Design Certification

- Relies on the regulations in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material,” for transportation package certification.
- Allows for the use of a risk methodology or other risk-informed approach for evaluating normal and/or accident conditions that has been endorsed or otherwise approved by the Commission to evaluate a package for certification (e.g., SECY-24-0062, “Risk-Informed Methodology for a Future Transportable Triso-Based Micro-Reactor Package Application” (ML23320A124))
- Includes provisions for the fueled reactor as the transportation package
- Includes provisions for the fueled reactor as the contents of a transportation package
- Requires that features to prevent criticality are in place during transportation

Subpart P – Operator Licensing and Human Factors

Generally Licensed Reactor Operators (GLRO)

- If no reliance on operator action to maintain the reactor below 1 rem dose-based entry criterion (10 CFR 57.25(a)), then a general license approach is permitted.
- Defined in Part 57 as an operator-independent facility.
- A general license is required to:
  - Manipulate reactor controls.
  - Direct the activities of other licensed operators.
  - Monitor plant conditions during reactivity or power level changes.
- Licensee Responsibilities
  - Incorporate human-system interface requirements into the plant design.
  - Develop and maintain a staffing plan.
  - Develop, implement, and maintain training, examination and proficiency programs for initial and continuous training using a systems approach to training.
- Designation of GLROs
  - Upon successful completion of an initial training and examination program, licensee may designate individuals to perform GLRO activities for a specific facility.
- Annual Reporting to the NRC
  - Submit list of GLROs to the NRC annually, including new additions and deletions since last report.
  - Supports inspection oversight.

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Subpart P – Operator Licensing and Human Factors

Specifically Licensed Reactor Operators Requirements

- If operator action is required to maintain the reactor below 1 rem dose-based entry criterion (10 CFR 57.25(a)), then a more focused operating licensing process is required.

- Defined in Part 57 as an operator-dependent facility.

- Key requirements include:
  - Individually licensed by the NRC (submission of applications)
  - Medical exams
  - NRC-approved licensing examination
  - Part 26 – “Fitness for Duty Programs”
  - NRC-approved training and proficiency/requalification
  - Subject to individual license conditions

Part 26 – Subpart P, Fitness for Duty

- If operator action is not relied upon to maintain the reactor below the 1 rem radiological dose criterion (10 CFR 57.25(a)) or a credible operator or maintenance error cannot result in exceeding the dose criterion:
  - The licensee can develop a fitness for duty (FFD) program of its own specification.
  - Otherwise, the licensee needs to implement proposed Subpart P of 10 CFR Part 26 (designed for Part 57 entities) *or* all of 10 CFR Part 26 except Subparts K and P (what the currently operating power reactor licensees implement).

- NRC will provide guidance on the essential elements that should be included as part of a licensee-developed FFD program.

- FFD program information is required to be included in the license application.

- Proposed Subpart P of 10 CFR Part 26 is structured after FFD program in 10 CFR Part 53 (Subpart M of Part 26) using the lowest tier FFD requirements.

- The requirements in proposed Subpart P of 10 CFR Part 26 are essentially equivalent to those requirements in 10 CFR Part 26, Subpart K, “FFD Program for Construction,” but have been supplemented by requirements from 10 CFR Part 26, Subparts A, “Administrative Provisions,” E, “Collecting Specimens for Testing,” I, “Managing Fatigue,” and O, “Inspection, Violations, and Penalties.”

Part 26 – Subpart P, Fitness for Duty

- 10 CFR Part 26, Subpart P would enable the use of:
  - Oral fluid, hair, or blood specimens for drug testing/screening
  - Split specimen collections
  - Virtual collection of specimens
  - Portable screening instruments
  - Live or captured video and audio for behavior observational monitoring
  - Offsite collection facilities

Part 57 Licensing Pathways

- Proposed Part 57 would provide applicants with several licensing options to fit their deployment models:
  - A joint application for a CP and associated OL(s) for deployment of nuclear reactors and nuclear plants.
  - An ML for approval and manufacture of nuclear reactors of standardized designs.
  - A standard design approval (SDA) for approval of entire reactor designs or major portions thereof.
  - Applicants would be able to combine requests for these types of licenses and approvals with requests for licenses, approvals, and certifications under other regulations (e.g., 10 CFR Parts 70, 71, and 72) in a single application to holistically address their deployment strategies.

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Part 57 Licensing Pathways

- Proposed Part 57 would include expanded finality provisions to focus “nth-of-a-kind” licensing on applicant- and site-specific information.
  - Standard reactor and plant designs approved through a CP and OL or ML.
  - Standard operational programs approved through a CP and OL or ML.
  - Postulated site parameters for reactor and plant designs.
- Proposed Part 57 would increase flexibility for making changes.
  - ML holders could make changes without prior NRC approval (similar to 10 CFR 50.59).
  - OL holders for manufactured reactors could implement approved license amendments to the ML without filing their own amendment requests.
  - ML holders and OL holders that reference the ML could combine license amendment applications.
  - Analysis of each departure, both individually and cumulatively, from the design characteristics, site parameters, terms and conditions, or approved design of the nuclear reactor, nuclear plant, or manufactured reactor would not require special circumstances justifications.

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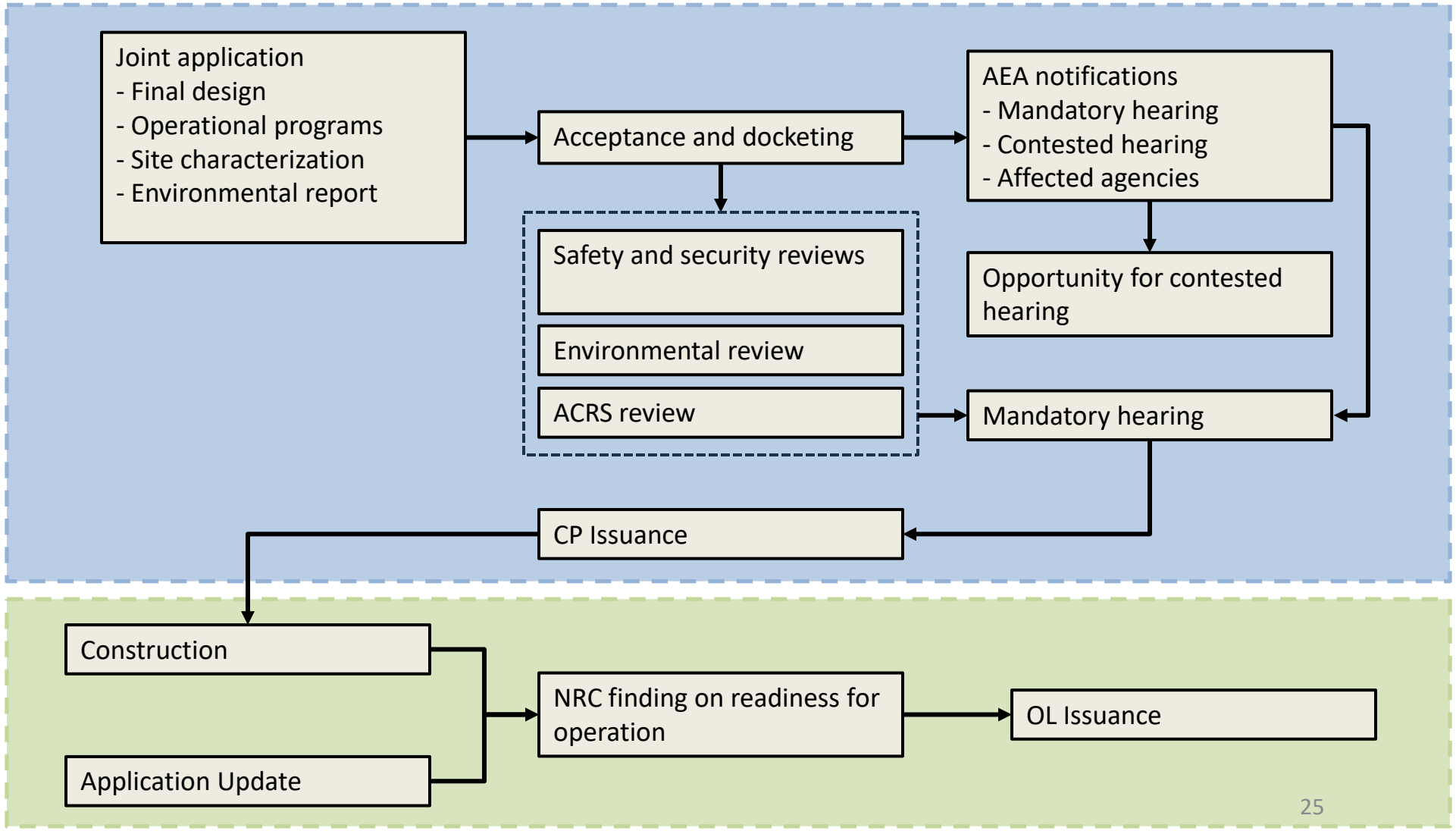
Part 57 Licensing Pathways

- Proposed Part 57 would include options for reactor site approvals in a CP
  - Approval of one site for a single reactor
  - Approval of multiple sites for a single reactor
  - Approval of multiple sites for multiple reactors
  - Approval of a large geographical area for one or more reactors
- Proposed Part 57 would establish a general license for construction
  - The general licensee must have a docketed joint application that references a Part 57 ML and a Part 57 CP and OL that were afforded generic finality.
  - The general licensee must have notified the NRC that all applicable permits, licenses, approvals, and other entitlements have been obtained, and all applicable Federal environmental consultations must have been completed.
  - The general licensee must not allow radioactive material, special nuclear material, or a manufactured reactor to be brought to the site of construction.
  - The activities undertaken by the general licensee are at its own risk.

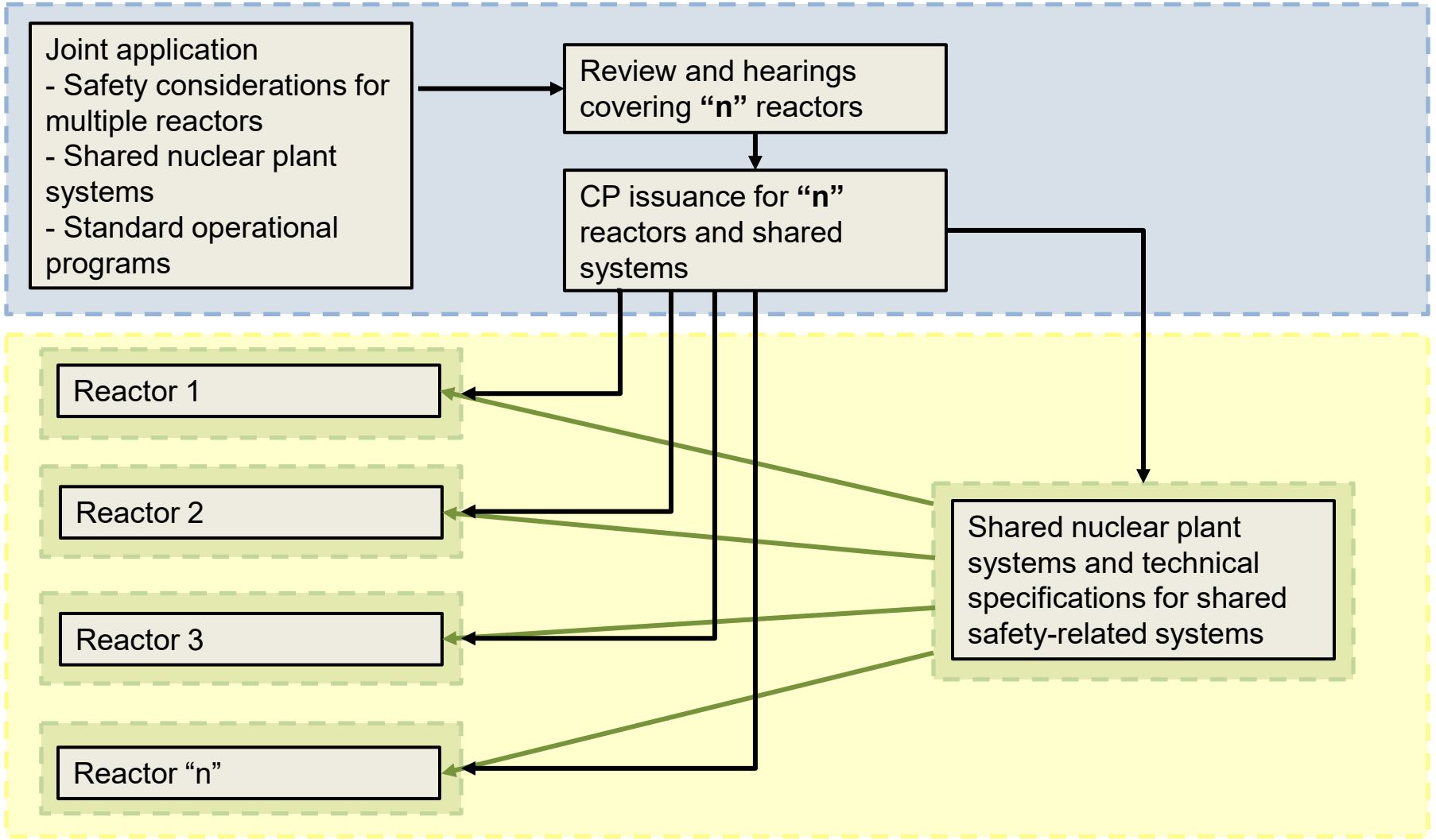
Part 57 Licensing Pathways

Example 1: Joint Application for a CP and OL for a Single Reactor



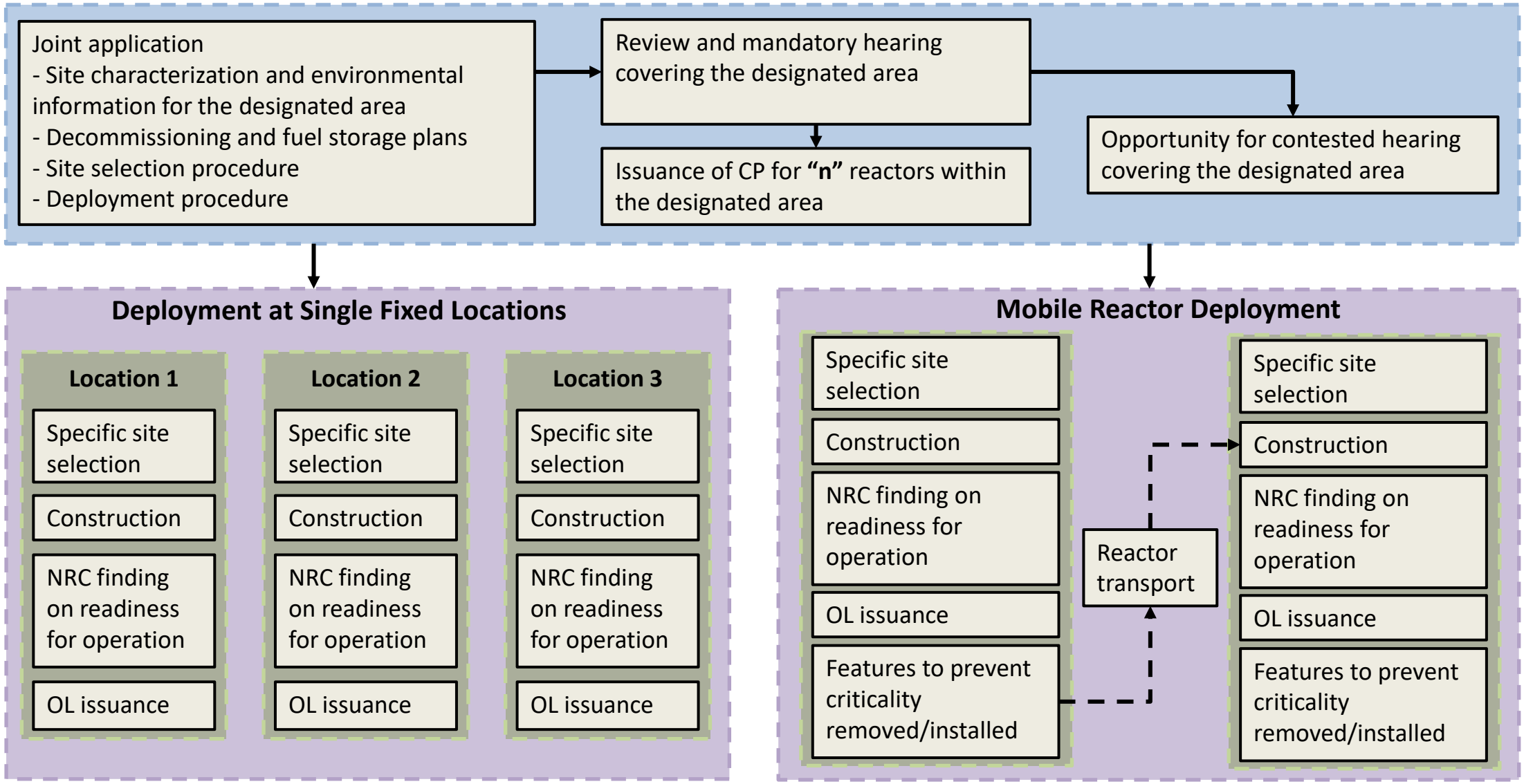
Part 57 Licensing Pathways

Example 2: Joint Application for a CP and Associated OLs for Multiple Reactors at a Single Nuclear Plant



Part 57 Licensing Pathways

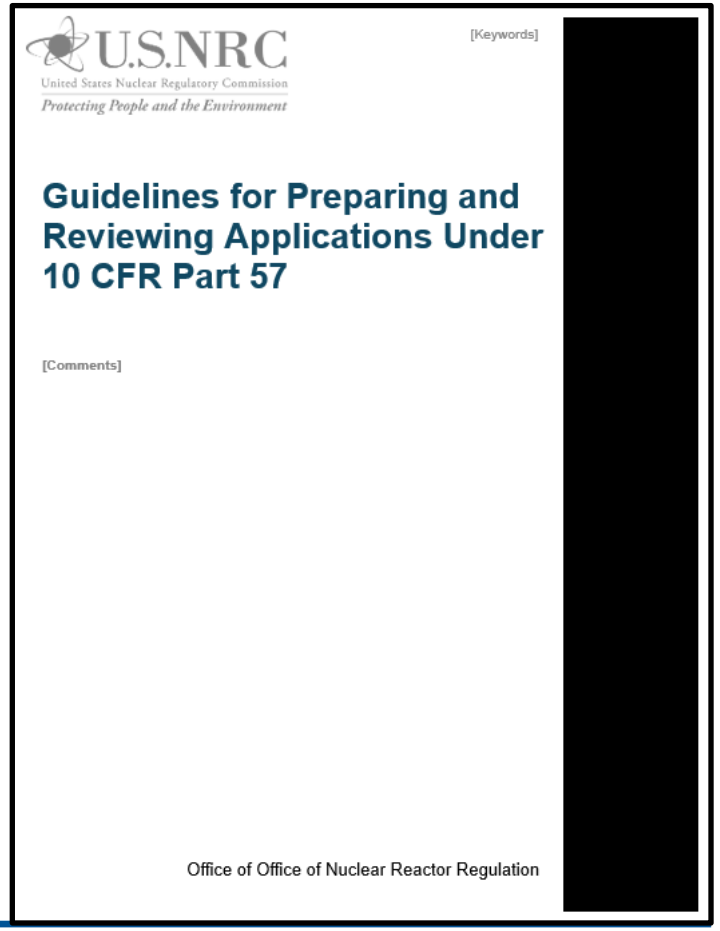
Example 3: Joint Application for a CP and Associated OLs for Reactors to be Deployed in a Large Designated Area



Part 57 Implementation Guidance

Draft NUREG-2271, “Guidelines for Preparing and Reviewing Applications Under 10 CFR Part 57”

- Supports meeting the Atomic Energy Act of 1954, as amended, for issuing a Section 103 commercial license
- Patterned from existing licensing practices (e.g., non-power production and utilization facilities) and guidance (NUREG-1537).
- Combines applicant and staff guidance into one document for developing and reviewing submittals under Part 57.
- Provides a flexible, graded hazard analysis approach consistent with DOE authorization methodologies.



Part 57 Implementation Guidance (cont.)

1. Facility Introduction
2. Site Characterization
3. **Design of Structures, Systems, and Components**
4. Reactor Description
5. Reactor Coolant and Cooling System
6. Engineered Safety Features
7. Instrumentation and Control Systems
8. Electrical Power Systems
9. Auxiliary Systems
10. Radiation Protection and Waste Management
11. Conduct of Operation and Operational Programs
12. **Accident Analysis**
13. Technical Specifications
14. Financial Qualifications
15. Decommissioning and Possession-Only License
16. Environmental Review
17. Part 57 Licensing Pathways
18. Operator Licensing and Human Factors Engineering

Appendix A: Safety-Related Terms for Advanced Nuclear Plants
Appendix B: Deterministic SSC Classification
Appendix C: Guidance and Information for Developing Advanced Reactor Source Terms
Appendix D: Format and Content of Decommissioning Plans for Facilities Licensed Under 10 CFR Part 57
Appendix E: Environmental Plant and Site Parameter Envelope Values
Appendix F: Graded Approach to Site Characterization for Part 57 Applications

Part 57 Guidance, Chapter 3, Design of Structures, Systems and Components

- **Safety-Focused Nuclear Design** - Emphasizes defense-in-depth, robust safety margins, and SSC features that prevent and mitigate uncontrolled radiological release for bounding design-basis accidents.
- **Advanced Reactor Expectations** - Incorporates passive and inherent safety characteristics, simplified systems, enhanced reliability, and extended operator grace periods.
- **Consistent Safety Terminology** - Uses Commission and internationally aligned terms—such as inherent safety characteristics, passive/active systems, fail-safe, and simplified safety systems—to improve clarity across technical and regulatory communities.
- **Comprehensive Principal Design Criteria (PDC)** - Classifies SSCs by safety significance and integrates the six required design attributes: reactivity control, heat removal, fission-product retention, shielding, effluent control, and security-by-design.
- **Hazard-Resilient Facility Design** - Ensures SSC performance under meteorological, hydrological, and seismic events through appropriate codes, standards, redundancy, and performance-based criteria.
- **Site Parameter Envelope (SPE)** - Establishes bounding environmental and geologic parameters to support standardized review, multi-site deployment, and predictable licensing outcomes.

Part 57 Guidance, Chapter 12, Accident Analysis

Consequence-Oriented Framework: Part 57 emphasizes outcomes by requiring applicants to establish the licensing basis through Chapter 12, Accident Analysis, ensuring safety performance is demonstrated rather than prescribing specific design features. (SRM-SECY-98-144 desire for risk-informed and performance-based regulation)

Graded Systematic Risk Evaluation: The rule adopts a risk-informed, graded approach, allowing the depth and rigor of analysis to scale with the potential consequences of accidents, promoting efficiency while maintaining safety. (Hazards Anaysis → Design-Basis Events → Design Basis Accidents → MHA or MCA)

Flexible Risk Analysis Methods: Applicants have the discretion to select the risk analysis methodology that best aligns with their design and business case, enabling innovation while meeting regulatory objectives. (10 CFR 57.60(a)(12) requires description of selected risk-analysis method)

Traditional Nuclear Safety Framework: Defense-in-depth and safety margins remain integral, focusing on safety-related SSCs required to manage design basis accidents and achieve safe shutdown conditions. This approach supports both prevention and mitigation strategies and informs the development of Technical Specifications. (10 CFR 57.3, “Definitions,” provides definitions of these terms)

Part 57 Guidance, Chapter 12, Accident Analysis (Cont.)

12.1 Background

12.2 Accident Analyses Methodology

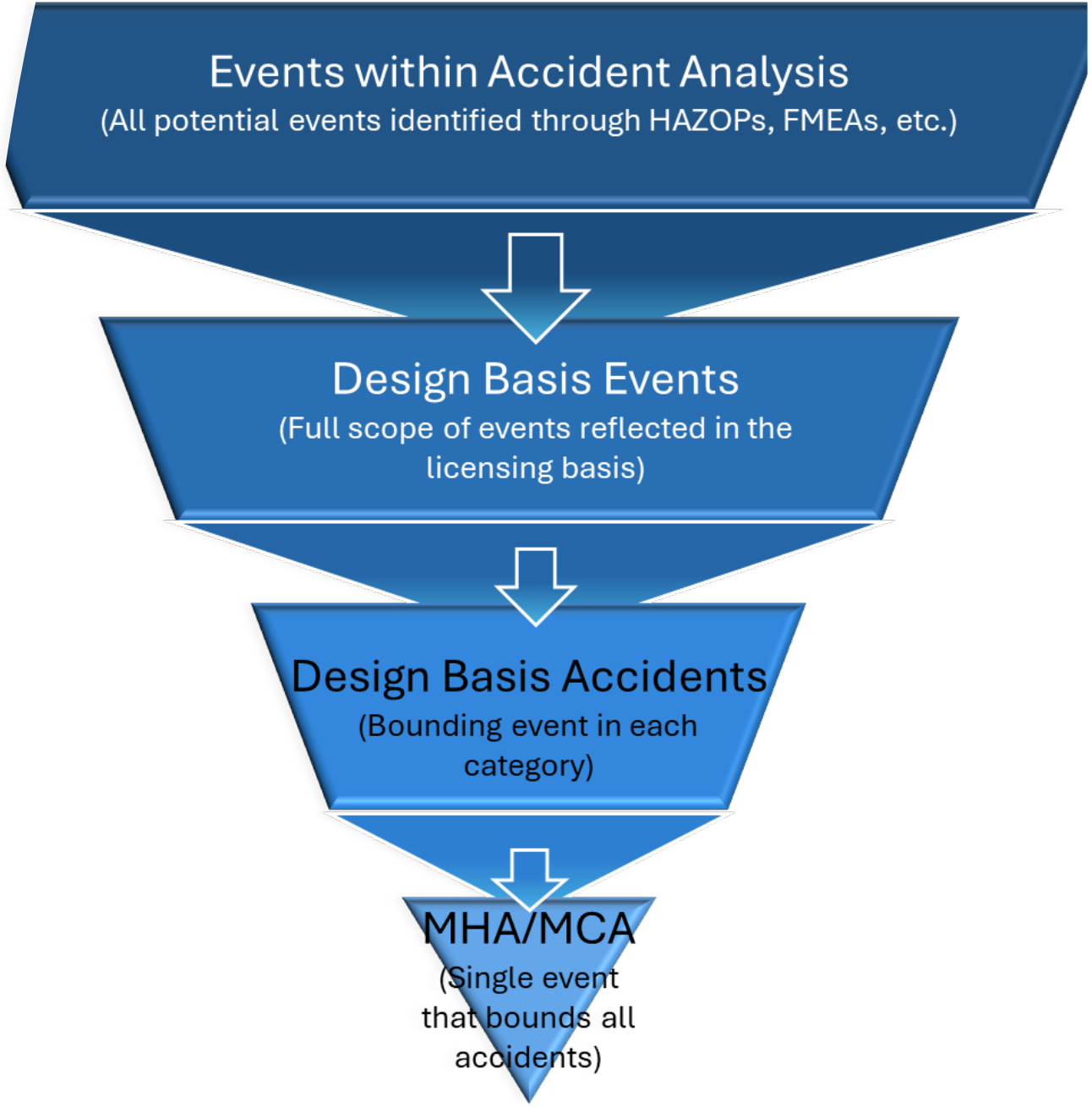
12.3 Credible Design-Basis Event Categorization

12.4 Design-Basis Accident Analyses and Consequences

12.5 Maximum Hypothetical Accident or Maximum Credible Accident

12.6 Defense-in-Depth Evaluation

12.7 Summary and Conclusions



From Figure 12-1 Relationship Between Event Groupings Within the Accident Analysis

Part 57 Guidance, Chapter 12, Accident Analysis (Cont.)

MHA and MCA

- Purpose: Siting the nuclear reactor or plant and sizing the site area boundary.
- Accident Dose Criterion: Applicants must show their reactor design meets the 1 rem TEDE limit using MHA or MCA methodologies.
- Bounding Consequence Analysis Assumptions: Bounds credible DBAs, assume a large source term from core damage, maximum leak rates, adverse dispersion, and an individual at the plume centerline without protection.
- Source Term Definition: Per 10 CFR 57.3, the source term includes radionuclide mix, physical/chemical form, release timing, and inventory fractions.

Descriptions of the MHA and MCA are as follows

Maximum Hypothetical Accident:

- A postulated accident scenario, often highly conservative, that assumes a severe release of radioactive material consistent with physical laws, regardless of probability.
- Does not necessarily reflect a realistic or credible sequence of events but instead represents a bounding case.
- Useful in cases to accept the uncertainty in system performance.

Maximum Credible Accident:

- A postulated accident scenario, often highly conservative, that assumes the most severe release of a source term composed of fission products consistent with physical laws, under credible accident conditions.
- Unlike the MHA, the MCA does not rely on physically unrealistic or excessively conservative assumptions, focusing instead on events that are credible given the technology, safety systems, and plant operating conditions.
- Used in facility-specific safety analyses to demonstrate that SSCs can perform their intended safety functions without exceeding accident dose acceptance criteria.

Part 57 Guidance, Chapter 12, Accident Analysis (Cont.)

More on ***Establishing the Site Boundary (12.5.1.1)***

- Site Boundary Complexity: Number of units, layout, and integration level drive safety analysis complexity.
- Per-Reactor Boundary: Define boundaries individually for MHA/MCA unless shared SSCs create credible multi-unit events.
- Multi-Unit Considerations: Maximize operational margin relative to the 1 rem TEDE criterion; additional flexibilities may be needed beyond initial siting assumptions.
- Provides several reactor and nuclear plant arrangement site boundary examples.

The Part 57 1-rem eligibility criterion is applied at the site level. From a careful reading of § 57.25(a) (see bold text):

> ”To be eligible for a construction permit and operating license or a manufacturing license under this part, an applicant must demonstrate that its **nuclear reactor or nuclear plant design and operation** meets the following entry criteria:
>
> (a) An evaluation of the applicable radiological consequences shows with reasonable assurance that any individual located in the unrestricted area following the onset of a postulated accident that bounds a broad range of design basis **accidents would not exceed 1 rem** (0.01 Sv) TEDE for the duration of the accident; …”

Therefore, a **nuclear plant** with several reactors is required to size the site boundary at a location that would not exceed 1 rem (0.01 Sv).

Part 57 Guidance, Chapter 12, Accident Analysis (Cont.)

- Single-Unit Configuration with Radial Simplification.
  - Site boundary is typically defined by a simple radial distance from the reactor center or the most conservative release point.
  - Licensee has total authority over all activities.

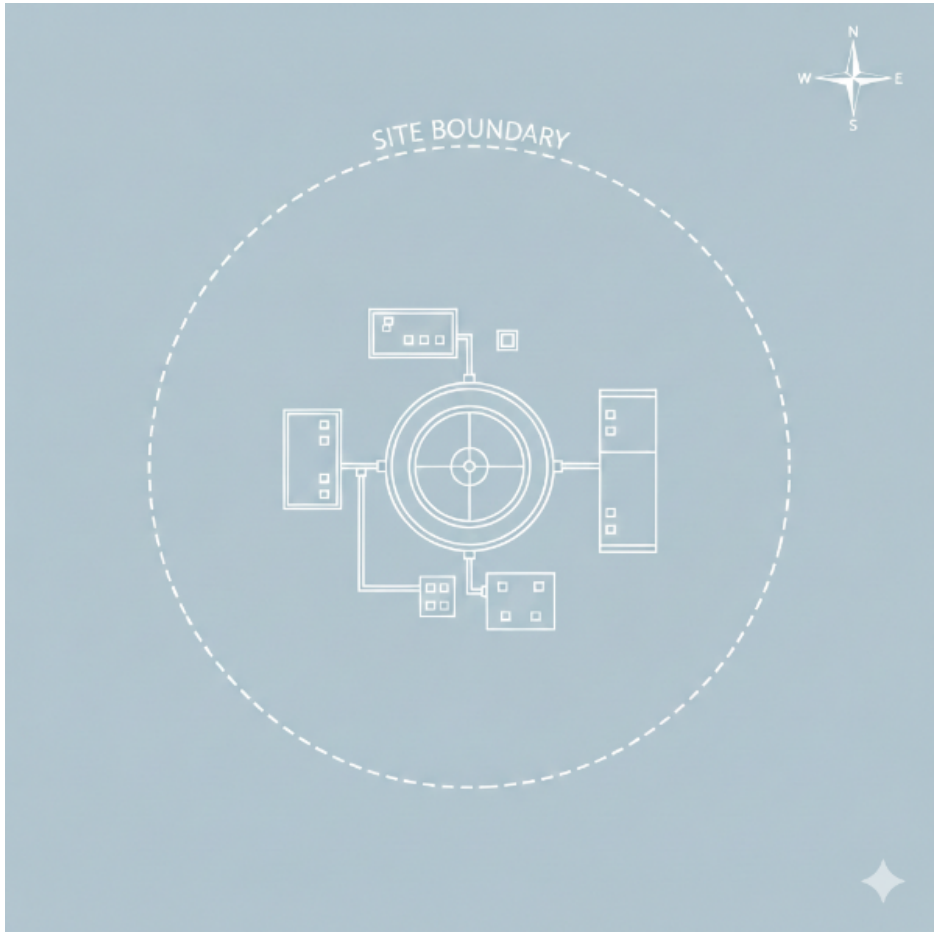


Figure 12-4 Schematic Representation of Single-Unit Configuration Relative to Site Boundary

Part 57 Guidance, Chapter 12, Accident Analysis (Cont.)

- Large Area with Multiple Reactors and Independent SSCs
  - Reactors may be treated as independent radiological sources.
  - Each reactor may have its own individual site boundary; however, these boundaries may be nested within a larger, licensee-controlled perimeter.
  - **Spatial Separation**: Units are sited far enough apart that the site boundary of “Unit A” does not overlap with the site boundary of “Unit B.”
  - **Operational Control**: While the reactors are physically separate, the licensee may manage the entire perimeter to simplify security and emergency planning.

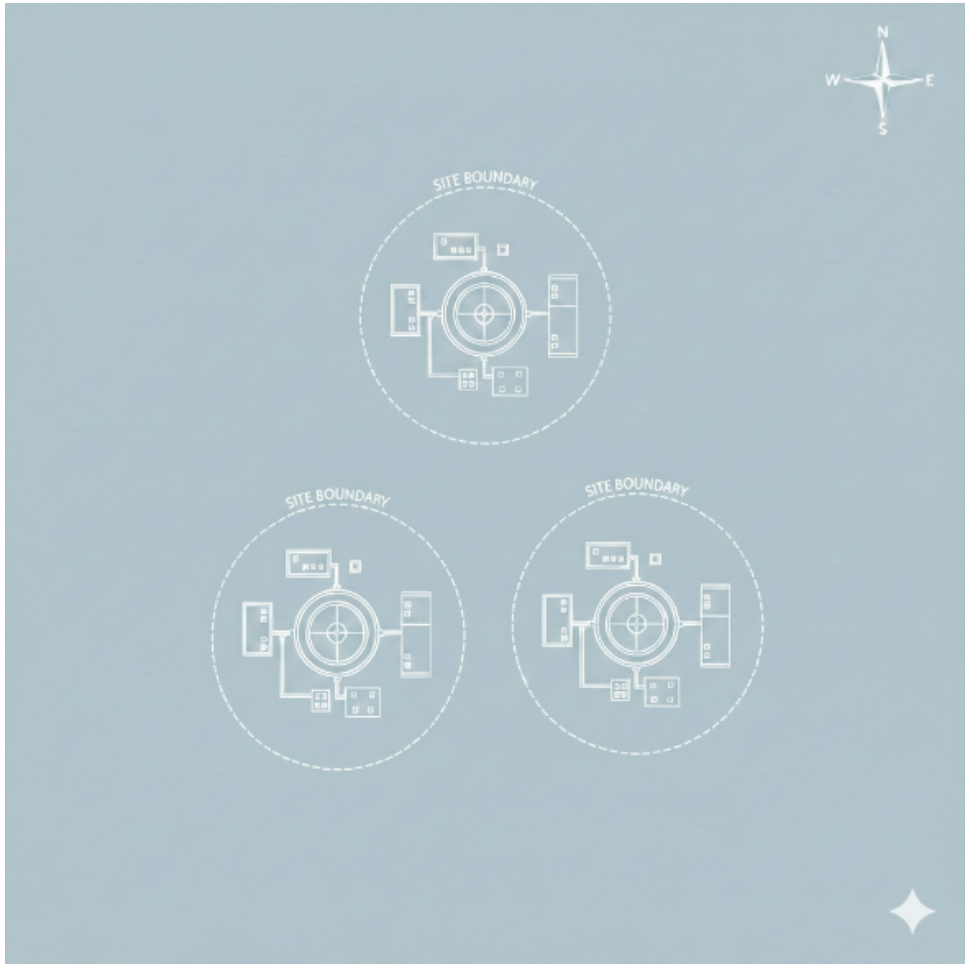


Figure 12-5 Schematic Representation of a Large Area Configuration with Independent SSCs Relative to Site Boundaries

Part 57 Guidance, Chapter 12, Accident Analysis (Cont.)

- Multi-Unit Nuclear Plant with Shared SSCs Meeting 1-rem Site Boundary
  - Reactors share safety-related SSCs-such as a common control room, shared emergency power, or a collective ultimate heat sink-the site boundary analysis becomes more rigorous.
  - **Bounding Source Term**: Because a failure in a shared system could potentially impact multiple reactors simultaneously, the applicant must establish a bounding MHC/MCA source term. This would involve calculating the credible DBA release from all affected units (see subsection 12.4.11) to ensure the site boundary.
  - **Integrated Boundary**: In this scenario, the site boundary is typically not a series of individual circles, but a single, contiguous perimeter encompassing all nuclear reactor units and their shared infrastructure.

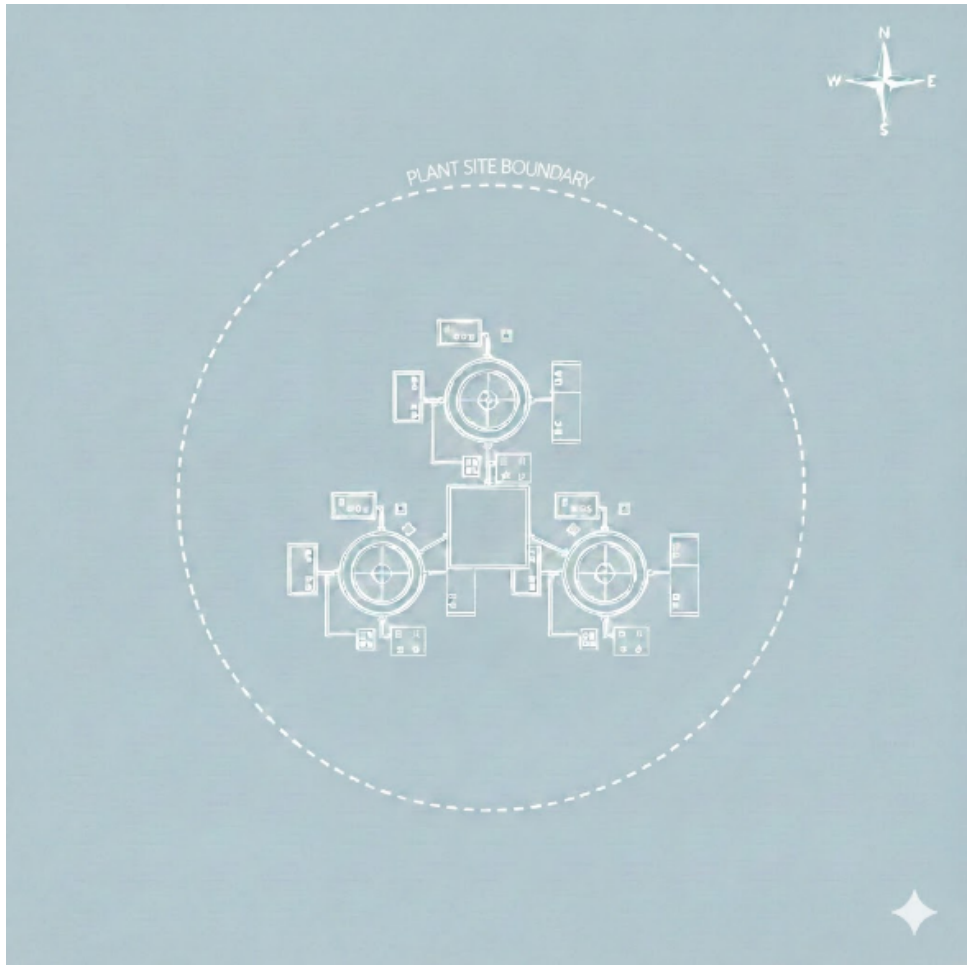


Figure 12-6 Schematic Representation of Multi-Unit Site with Shared SSCs Relative to Site Boundary

Part 57 Guidance, Chapter 12, Accident Analysis (Cont.)

- Multi-Unit Nuclear Plant Meeting 1-rem Site Boundary:
  - Assess the bounding credible DBA for each reactor that can impact all reactors regardless of safety-related SSCs being shared or not. The likely credible DBA that impacts all reactors may be a seismic event.
  - Develop a nuclear **plant-based MHA/MCA source term** that bounds the nuclear plant credible DBA impacting all reactors at the nuclear plant.
  - Treat the MHA/MCA release as a point release near or at the geometric center of the nuclear plant.

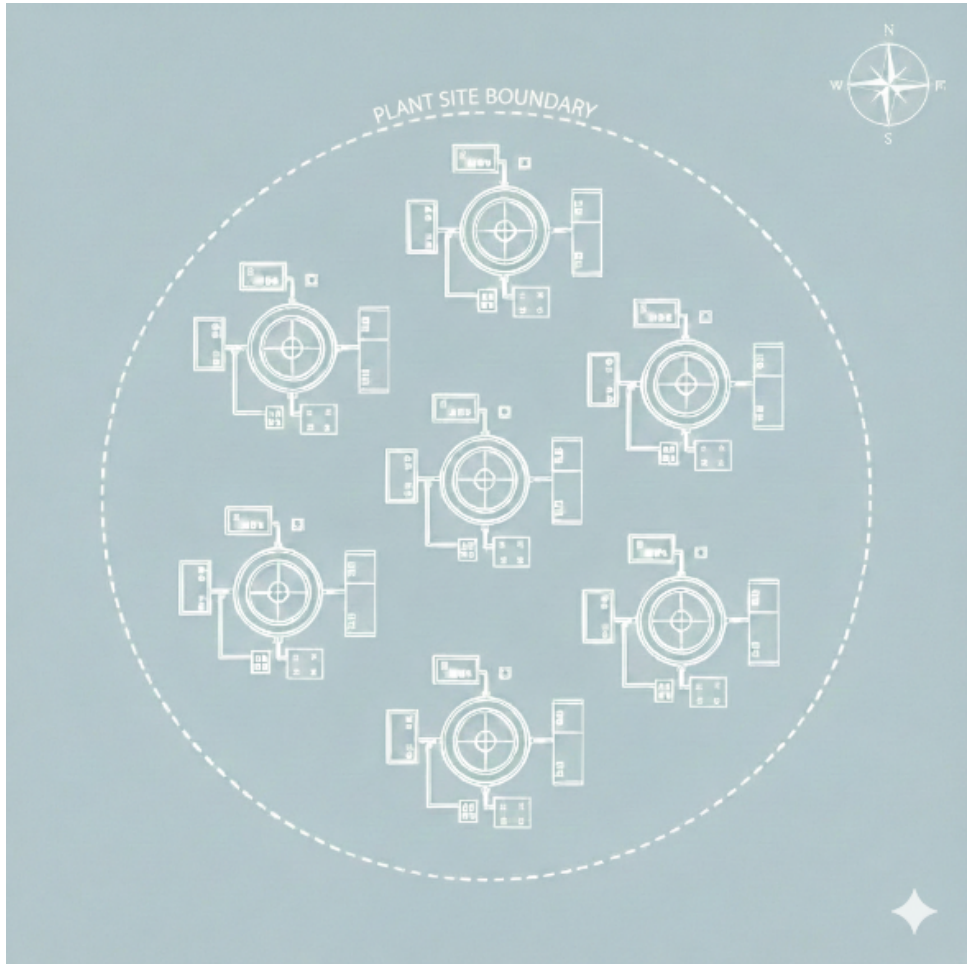


Figure 12-6 Schematic Representation of Multi-Unit Site with Shared SSCs Relative to Site Boundary

Schedule

- Public comment period – 45 days (ends June 15, 2026)
- Public meeting – May 2026
- Estimated date to publish final rule in the Federal Register: December 1, 2026

Background Slides

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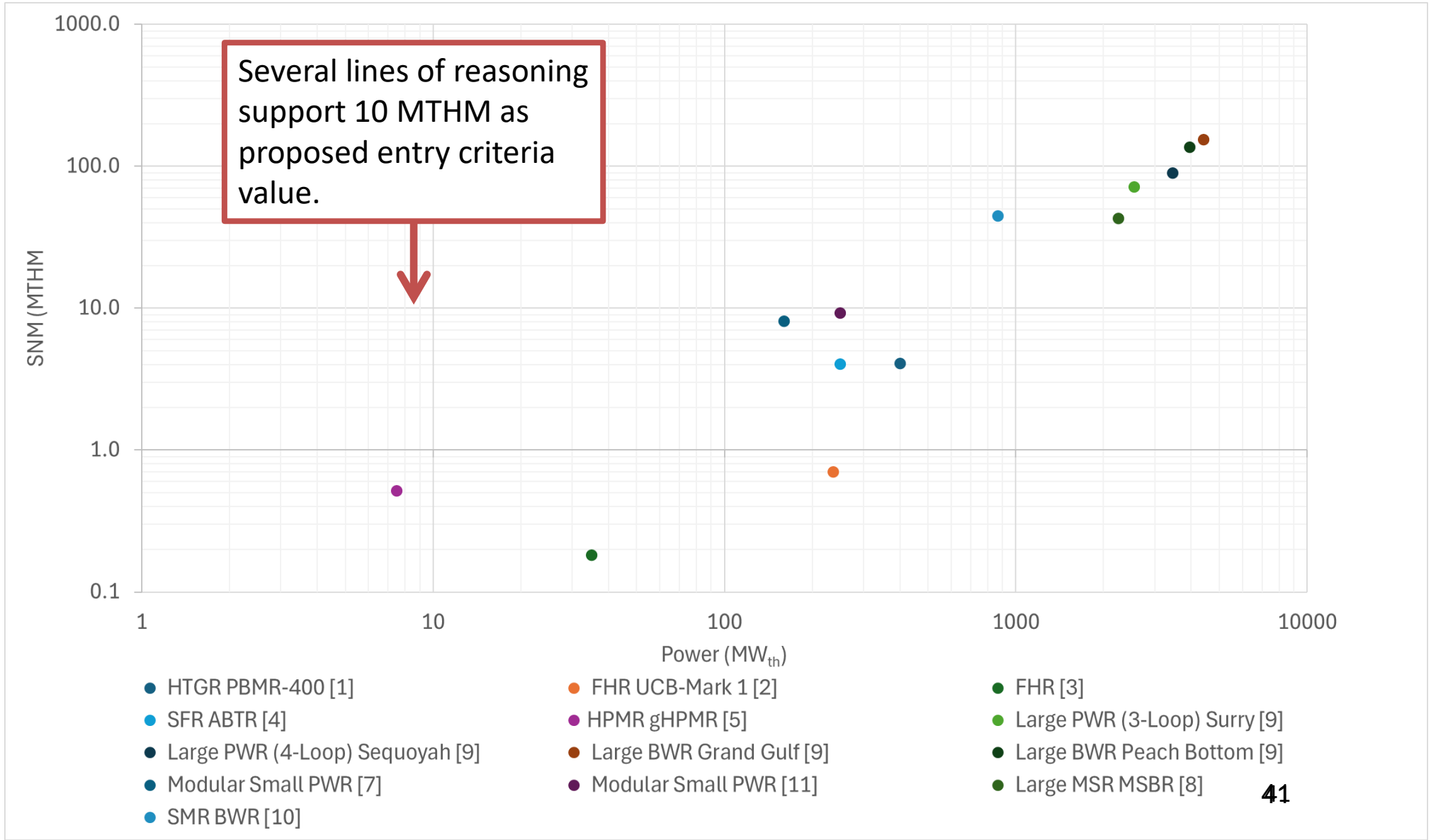
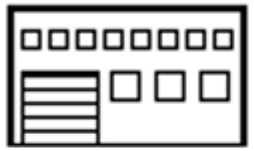
Fuel Mass Limit



Conceptual Deployment Model for Transportable Microreactors

Factory or Manufacturing Facility



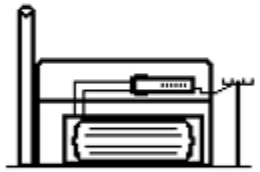
Fabricate the reactor, load fuel, and potentially operate the reactor for functional testing

Transportation to the Deployment Site

Reactors may contain fresh or irradiated fuel

Deployment Site – Power Operation

Stand-alone, self-contained microreactor design

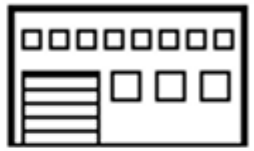


Core module with onsite reactor building and power conversion equipment

Transportation from the Deployment Site

Reactors may contain spent or irradiated fuel

Decommissioning or Refurbishing Facility



Remove fuel and decommission the reactor, recycle components and systems, or refurbish and refuel the reactor for redeployment

Redeployment

Deployment Lifecycle

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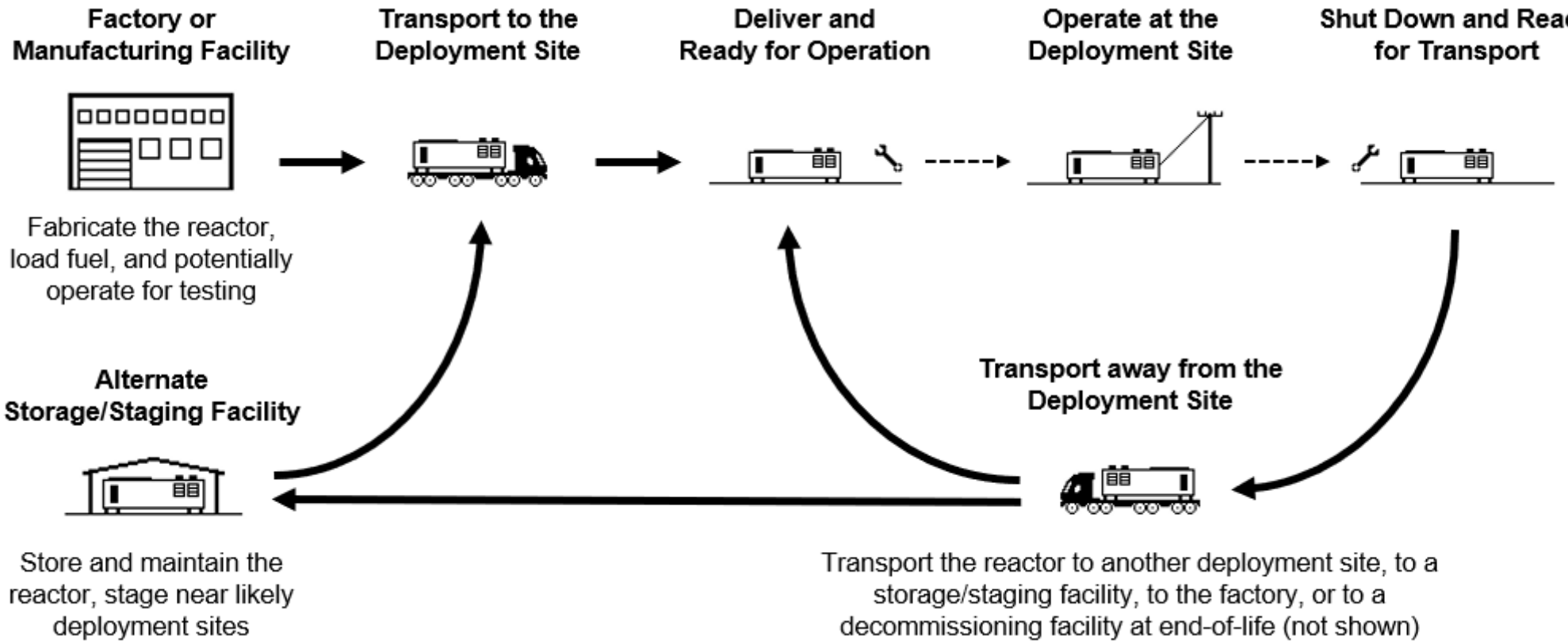
Conceptual Deployment Model for Mobile Microreactors*



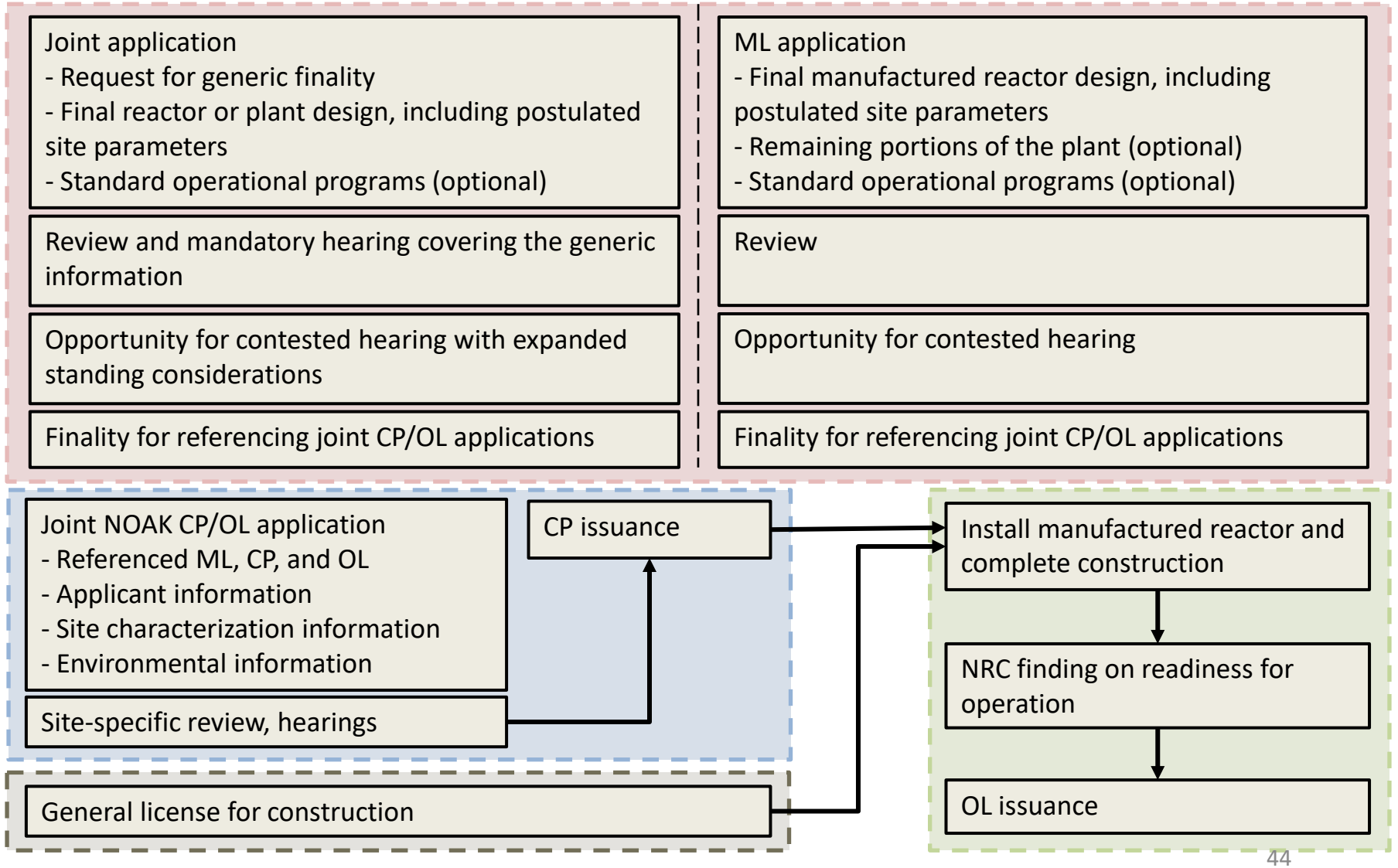
Indicates removal of features to prevent criticality

Indicates installation of features to prevent criticality

* A mobile microreactor would be a microreactor that is intended to be operated at more than one fixed location. Mobile microreactors would not be authorized to operate while in motion.

Proposed Part 57 Licensing Pathways

Example 4: Finality and "Nth-of-a-Kind" (NOAK) Licensing and Deployment



Transformative Aspects of Part 57 (cont'd)

Specific provisions include:

- Ability to provide custom definitions without requesting exemptions: § 57.3
- Redefine *construction* to just safety-related SSCs: § 57.3
- Finality of Part 50 and 52 references in Part 57 applications: § 57.16(b)(1)-(2),
- Finality of operational programs in a ML or first-of-a-kind CP/OL: § 57.16(b)(1)
- Finality of Part 57 MLs, CPs, and OLs and departures in referencing applications: § 57.16(c)
- Combining applications for all aspects of microreactor deployment: § 57.18(a)
- Reduce regulatory burden of changes affecting manufacturing licenses and manufactured reactors: §§ 57.18(a)(4), 57.175(d), 57.312(a), 57.312(b) [implement amendments to the ML without an amendment to the OL]
- Licensing multi-reactor plants, including replacement reactors and operational testing at a manufacturing facility: § 57.18(a)(5)
- Fleet deployment and rapid deployment: § 57.18(a)(2) [multiple sites] and § 57.60(d) [large areas]
- Referencing DOE and DoD authorizations: § 57.18(e)
- Joint application for CP and OL: § 57.19(f)

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Transformative Aspects of Part 57 (cont'd)

Specific provisions include:

- Dose-based eligibility criterion: § 57.25
- Design criteria attributes: § 57.30
- General license for construction: § 57.45(d)
- Remote monitoring, remote operation, and autonomous operation: § 57.60(a)(1)(iv)-(vi) and (a)(8)(xi)
- Applicant justified quality assurance program: § 57.60(a)(3)
- Siting and graded site characterization: § 57.60(a)(5)
- Use of postulated site parameter envelopes: §§ 57.60(a)(5), 57.155(b), and 57.350(b)(1)
- Appropriate emergency preparedness and security plans: § 57.60(a)(8)(iv) and (a)(8)(v)
- Custom use of codes and standards: § 57.60(a)(9)
- Increased enrichment of fuel for fresh fuel storage safety measures: § 57.60(a)(13)
- Categorical exclusion for environmental review for CPs and OLs: § 57.60(b)

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Transformative Aspects of Part 57 (cont'd)

Specific provisions include:

- Standardized, generic operational programs: §§ 57.60(c), 57.142(e), 57.160(a), 57.172(a)(6), 57.175(c)
- Reduce regulatory burden of changes affecting finality of CP/OLs and MLs: §§ 57.142(c), 57.175(d)
- Approval of the complete plant design through an ML: §§ 57.155(c)(8), 57.172(a)(3), 57.172(b)(4), 57.175(c)
- Fuel loading at a manufacturing facility and transportation of fueled reactors: §§ 57.160(f), 57.197(d), 57.197(e)
- Irradiated fuel storage by OL holders, ML holders, and specific licensees: § 57.300
- Flexible decommissioning based on non-power reactor regulations, option for up-front approval, and site-specific cost estimates: §§ 57.60(a)(8)(xvii), 57.305
- Transportation package design certification, including the reactor as a transportation package: Subpart I
- Use of generally licensed reactor operators: Subpart P
- Use of Fitness for Duty program of licensee's own specification: § 26.3(f)

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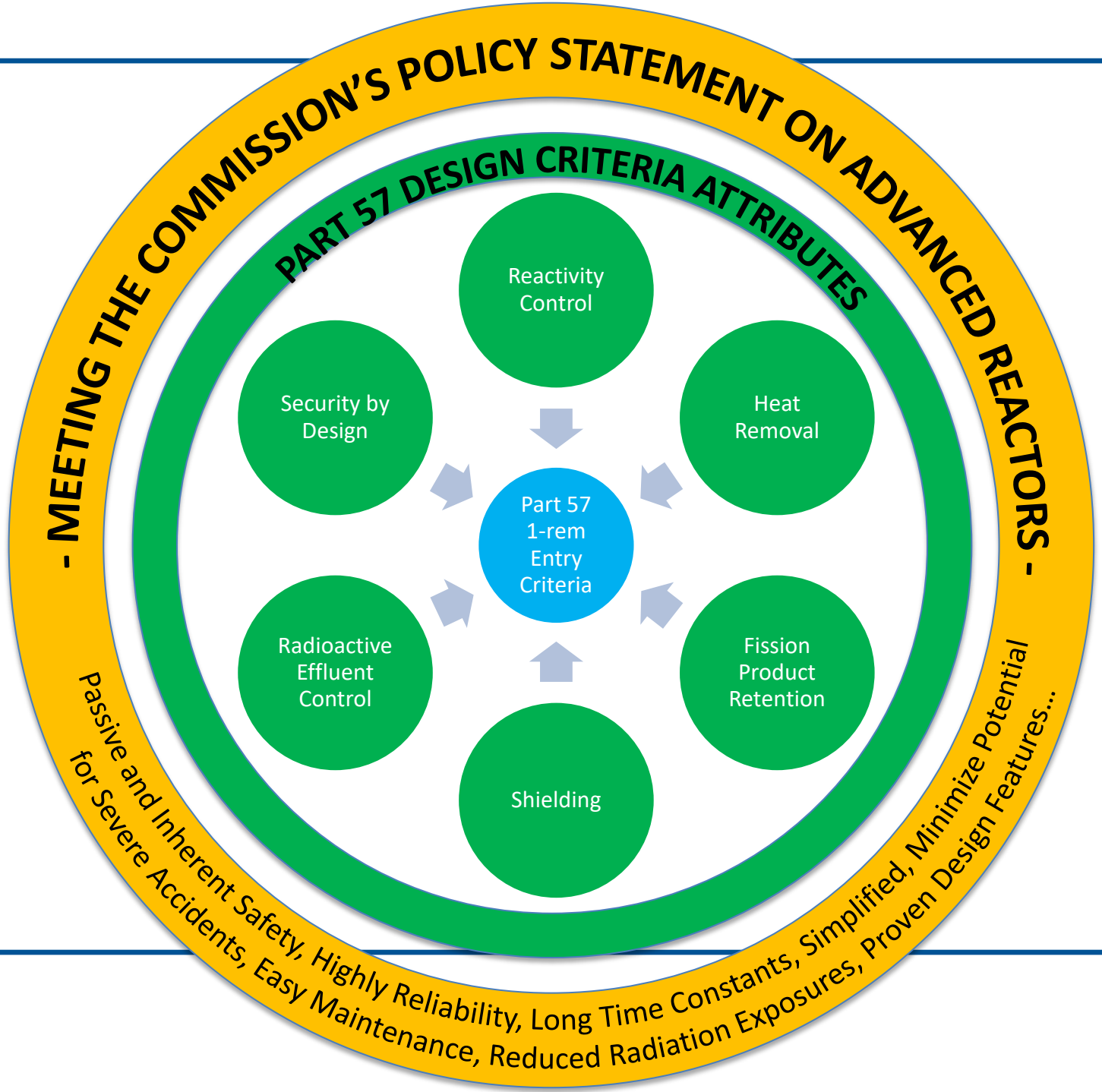

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