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

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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REGULATORY RULEMAKING, POLICIES AND PRACTICES

SUBCOMMITTEE

+ + + + +

TUESDAY

DECEMBER 17, 2024

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The Subcommittee met via Videoconference, at 8:30 a.m. EST, Ronald G. Ballinger, Chairman, presiding.

SUBCOMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman

VICKI M. BIER, Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

CRAIG D. HARRINGTON, Member

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ROBERT P. MARTIN, Member

SCOTT P. PALMTAG, Member

DAVID A. PETTI, Member

THOMAS E. ROBERTS, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:
DENNIS BLEY

DESIGNATED FEDERAL OFFICIAL:
ZENA ABDULLAHI
WEIDONG WANG

ALSO PRESENT:
PHILIP BENAVIDES, NMSS/REFS/RRPB
KRISTY BUCHOLTZ, NRR/DRA/APOB
LARRY BURKHART, ACRS/TSB
THERESA CLARK, NRR/DSS
JAMES CORSON, RES/DSA/FSCB
SE-KWON JUNG, NRR/DEX
JOHN LEHNING, NRR/DSS
JOSEPH MESSINA, NRR/DSS/SFNB
CHARLEY PEABODY, NRR/DSS/SNSB
JASON PIOTTER, NMSS/DFM/NF
DAVE RUDLAND, NRR/DNRL
ROBERT TREGONING, RES/DE

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

8:30 a.m.

CHAIR BALLINGER: The meeting will now come to order. This is a meeting of the Fuels, Materials, and Structure Subcommittee of the Advisory Committee on Reactor Safeguards. I'm Ron Ballinger, chair of today's subcommittee meeting.

ACR members in attendance in person are myself, Craig Harrington, Tom Roberts, Dave Petti, Bob Martin, and Scott Palmtag. Members that are in attendance virtually or may be or should be virtually by Teams are Walt Kirchner, Greg Halnon, Matt Sunseri, Vicki Bier, and Vesna Dimitrijevic. We have one consulting participant virtually, and I believe that's Dennis Bley, but I can't see. He's there. So I probably missed somebody.

Zena Abdullahi and Weidong Wang of ACRS staff are the Designated Federal Officers for this meeting. No member conflicts of interest were identified for today's meeting. We have a quorum.

During today's meeting, the subcommittee will receive a presentation on technical topics regarding increased enrichment of conventional and accident tolerant fuel designs for light-water reactor rulemaking. The U.S. Nuclear Regulatory Commission is proposing to amend its regulation related to the use of conventional and accident tolerant light-water reactor fuel designs. The NRC's goal is to establish effective and efficient licensing of the use of fuels enriched to greater than 5 weight percent uranium-235 while continuing to provide reasonable assurance of adequate protection of public health and safety. The new requirements also would address fuel fragmentation, relocation, and dispersal, FFRD, in relation to the key accident tolerant fuel components of increased enrichment and burnup limits.

The ACRS was established by statute and is governed by the Federal Advisory Committee Act, or FACA. The NRC implements FACA in accordance with our regulations. Per these regulations and the committee's bylaws, the ACRS speaks only through its published letter reports. All member comments should be regarded as only the individual opinion of that member, not a committee position.

All relevant information related to ACRS activities, such as letters, rules for meeting participation, and transcripts are located on the NRC public website and can be easily found by typing about us ACRS in the search field on the NRC home page. The ACRS, consistent with the agency's value of public transparency in regulation of nuclear facilities provides opportunity for public input and comment during our proceedings. We have received no written statements or requests to make an oral statement from the public. We also set aside time at the end of this meeting for public comments.

The ACRS will gather information, analyze relevant issues and facts, and formulate proposed conclusions and recommendations, as appropriate, for deliberation by the full committee. A transcript of the meeting is being kept and will be posted on our website.

When addressing the subcommittee, the participants should first identify themselves and speak with sufficient clarity and volume so that they may be readily heard. You also make the stenographer if you do that. If you are not speaking, please mute your computer on Teams or by pressing star six if you're on a phone. Please do not use the Teams chat feature to conduct sidebar discussions related to the presentations. Rather, limit use of the meeting chat function to report IT problems. For anyone in the room, please put all your electronic devices in silent mode. Hopefully, I did that. Yes. 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 and they're pretty sensitive.

For presenters, your table microphones are unidirectional. I think most of you know that. You have to really get close to them. Finally, if you have any feedback for the ACRS about today's meeting, we encourage you to fill out the public meeting feedback form on the NRC's website.

We'll now proceed with the meeting, and, Philip, are you going to give some opening remarks?

MR. BENAVIDES: Thank you. I'm Phil Beneavides, project manager in Office of Nuclear Material Safety and Safeguards, and I'm the rulemaking project manager in charge of the rulemaking on increased enrichment in conventional and accident tolerant fuel designs for light-water reactors. Today, we're going to present the draft proposed rule for the increased enrichment rule, proposed rule. We will begin with opening remarks from Theresa Clark, the director of the Division of Safety Systems, followed by a brief overview and status of the increased enrichment rulemaking which will lead into presentations from subject matter experts for each technical topic.

Next slide. Theresa.

MS. CLARK: Thanks, Phil. And good morning to all the members. I'm really pleased to see all of you, as well as the significant audience here, so this is obviously a topic of high interest. It's of strong interest to us, as well. I'm Theresa Clark again. I'm the director of the Division of Safety Systems here in the Office of Nuclear Reactor Regulation at the NRC. I'm also the chair of the Management Steering Committee for this rulemaking, so very heavily invested in this rule. And I just want to say I really appreciate the committee devoting so much of its time to this rulemaking, as well as to the related discussions on Reg Guide 1.183. I know it's a lot of work. It's been a lot of work for our collective team, and I want to appreciate their efforts, as well. And special shoutouts to Phil here, as well as to Joseph Messina, who is the lead in my division, and you'll be hearing a lot from him. He's been coordinating with the rest of the team to keep this rule moving.

One of the reasons I'm especially proud of this rulemaking is because it's collaborative, creative, and groundbreaking. We figured out new ways to integrate feedback on the rule while still keeping to our schedule, and the result is expected to be a big enabler to advancements in both safety and economics across the industry.

Over the next couple of days, you'll hear a lot of about FFRD, which was already defined: fuel fragmentation, relocation, and dispersal. We'll hear even more about what that actually is. And we began this rulemaking in 2022 with Commission direction to address that phenomenon within the rule, and the committee has been very interested, obviously, in our approach. It adds technical complexity to the rule for sure, but it's also an opportunity for us to unify several previously worked on 5046 rulemakings in a holistic and risk-informed way.

So by selecting, as you'll hear, FFRD Alternative 2 from the regulatory basis, we could leverage the extensive work that was done several years ago on the 5046(a) rulemaking. That helped us stay on schedule while repeating the benefits that licensees may be able to see of risk-informing the large-break LOCA analytical approach. Further, by integrating performance-based requirements into the rule, as you'll hear a lot about later, we could preserve flexibility for other ways that industry may want to address FFRD while resolving the safety issues with hydrogen embrittlement that were worked on previously in the 5046(c) rulemaking, and that was worked on by the staff and reviewed by the committee like 5046(a) was. So integrating these two efforts helped us stay on schedule and it will likely help your review, as well. At the same time, we were mindful of previously proposed requirements that may be overcome by events or more prescriptive than we need today, and so we thought hard about what's truly necessary as we shape this rule.

And you'll hear about lots of topics, other than FFRD, that I won't get into because Phil will do that overview, but I really just want to thank the committee for their effort in reviewing the documents and your flexibility in giving us multiple presentation time slots. That helps us a lot because the documents are still in their final review and concurrence stage. Phasing this and letting us present in January lets us make even more progress that way. So, for example, you'll see in some of the slides that some of the provisions have even been changed or deleted since we first sent you the package. That's a natural evolution of that concurrence process. And between this meeting and the January meeting, we'll be poised to give you the full picture of the rule.

CHAIR BALLINGER: So that means I have a question. The documents that we now have, at what point will we get the final documents?

MS. CLARK: Phil might answer that question even better than I would, but before the January meeting I think.

MR. BENAVIDES: Yes, it would have to be before the January meeting. I know, typically, you want 30 days before. We're still going through concurrence, and, yes, we will provide anything that's significant changes before the --

CHAIR BALLINGER: Any letter that we would write or will write would have to be based on --

MR. BENAVIDES: We want to have the letter reflect what's actually going up to the Commission. Correct.

MS. CLARK: And just so that I don't scare you too much, the sorts of things that you'll see are so far relatively minor changes related to reporting requirements or updates frequencies, those sorts of things where we've been re-weighing the burden, so not a major technical change so far, nor do I expect one.

CHAIR BALLINGER: Also, one last thing for the record. I'm told by certain moles within your organization that this whole process took a year, which is remarkable and, to my mind, sets a precedent for something that's as complex as this in the future. Thank you.

MS. CLARK: Thanks for that recognition. Couldn't have done it without a great team and without those prior efforts on our prior goals. So take it away, Phil. Thanks.

CHAIR BALLINGER: Okay. We now have a binary distribution here. I could understand her very, very well, but you're -- can you get a little closer to the --

MR. BENAVIDES: Oh, sure. Is that better?

CHAIR BALLINGER: Yes.

MR. BENAVIDES: All right. Thank you, Ron. With that, I'm going to provide an overview of the increased enrichment rulemaking. Next slide.

As a way to provide background on how we got to this point, I'd like to go back to the beginning when the issue was identified. Throughout the last few years, staff has seen increased enrichment interest from the industry for the use of fuel enriched above 5 weight percent uranium-235 or U-235. The NRC noted that although the current regulatory framework allows a licensee fuels above 5 weight percent U-235, this use of this fuel may result in new redemption requests for licensees. So as a proposed solution, NRC staff began pursuing rulemaking rather than licensing individual exemptions.

In December 2021, the staff provided the Commission with SECY-21-0109 requesting approval to begin the rulemaking process, which was approved, you know, in March of 2022.

Next slide. And when the Commission approval was granted in SRM SECY-21-0109 in March of 2022, the Commission specified several considerations to evaluate in addition to what was specified in the rulemaking plan. One was that the rule should only apply to high-assay low-enriched uranium levels. This was done for both nonproliferation and safeguards reasons and for the staff to focus on the range of enrichment most likely to be contemplated in future applications. In addition, staff was directed to address fuel fragmentation, relocation, and dispersal, and to take risk-informed approaches.

Next slide, please. The NRC issued the regulatory basis on September 8th, 2023. Stakeholder involvement throughout the process included public meetings which were held before the regulatory basis was issued on June 22nd of 2022 and after the regulatory basis was issued October 25th of 2023. The regulatory basis comment period was open from September 8th, 2023 through January 22nd, 2024.

In addition to the typical rulemaking engagement, staff shared fuel dispersal insights at the NRC's annual higher burnup workshop on September 3rd, 2024. The proposed rule is currently due to the Commission in March of 2025.

Next slide. This slide shows an overview of the rulemaking activity. We're still in the proposed rule package development stage, and we're currently working out way towards submission to the Commission in March of 2025. After the Commission reviews and approves the proposed rule package, the staff will finalize the proposed rule based on Commission direction. A Federal Registry notice will then be issued opening up the proposed rule for public comment in the purple box which is estimated to begin in July of next year, 2025.

After the public comment period closes, the staff will develop the draft final rule package which is expected to be sent to the Commission in September of 2026. Note that the dates listed are estimates and subject to change.

I do want to point out, in addition to

these ACRS engagements between now and February 2025 that we have plans. The staff plans to present these to ACRS towards the end of the final rule development, as well, prior to the draft final rule being sent to the Commission for consideration.

Next slide. The increased enrichment rulemaking addresses the following topics. Note that the topics highlighted in red will be presented over the next two days. These include criticality accident requirements and 10 CFR 50.68, packaging requirements for fissile material transportation in 10 CFR 71.55, control room design requirements in 10 CFR 50.67 and GDC-19, and also fuel fragmentation, relocation, and dispersal.

Next slide. This slides shows the associated guidance developed for the increased enrichment rulemaking with one draft guide associated with control and design requirements and six draft guides associated with fuel fragmentation, relocation, and dispersal. Please note that DG-1428 titled Plant-Specific Applicability of Transition Break Size will be presented during the January 2025 ACRS subcommittee meeting. The other draft guides will be discussed over the next few days.

Next slide. This slide shows the order of presenters for today. After this presentation, we'll transition to presentations on critical accident requirements and fissile packaging requirements. And then, after a short break, we'll transition to the fuel fragmentation, relocation, and dispersal presentations.

Next slide. And day two will include a continuation of the fuel fragmentation, relocation, and dispersal topic presentations in the morning, and then we'll transition to control room design criteria presentations in the afternoon.

Next slide. With that, any questions?

MEMBER PETTI: I just had a process curiosity question. Why is NMSS the lead? I mean, I understand enrichment, but if you look at all the people, just looking at the list, you think NRR should have something to do with it.

MR. BENAVIDES: Yes. Historically, rulemaking used to be -- I was in different offices when they created the Center of Expertise for rulemaking. It happened to be housed under the office of NMSS, you know. So if you look at our office, there's a materials rulemaking branch and a reactor rulemaking branch.

MEMBER PETTI: Thanks.

MR. BENAVIDES: You're welcome. Okay. Well, hearing no other questions, we can move on to the technical topic presentations. Thank you for your time. We're going to start with Charlie Peabody.

MR. PEABODY: Sorry. I'm attending remotely today, but I do appreciate the time to present to the committee on this topic. Aaron, can you advance to the next slide?

All right. So 50.68 was implemented in 1998, and that was basically to codify past exemptions to 70.24, which the issue with that was that active criticality monitoring and emergency preparedness requirements were required to ensure that you could respond to a criticality event in a spent fuel pool at a light-water reactor. So provide alternative compliance to this, we implemented k-effective safety limits in Parts (b)(2), (3), and (4) of this rule to remove the active criticality monitoring and emergency drill requirements if you could demonstrate that the k-effective in the pool would not exceed 0.95 at 95-percent probability and 95-percent confidence.

There is another provision in the rule, Paragraph (b)(7), which limited application to enrichments of less than or equal to 5 percent by weight U-235. Recently, we were approached by the industry with a desire to increase enrichments beyond 5 percent specifically for PWRs who wanted to have a 24-month cycle.

Next slide, please. So we've actually kind of started this a little bit. There was a license amendment from Vogtle to utilize lead test assemblies above 5 percent enrichment that we've approved that included an exemption to 50.68 Paragraph (b)(7), and we're forecasting that batch cycles would be implemented for that potentially as early as the end of the 2020 decade.

We also, over the past year, we completed a research study that we contracted through Oakridge National Labs. I have the number of the study, as well as the accession number. It is publicly available if anybody wants to review it. Basically, what the study showed is that it actually exceeded our expectations in terms of utilizing existing integrated fuel burner absorber and gadolinium absorber technologies to, you know -- we think that if we can use them that we can possibly get criticality down. Even in some hypothetical cases where we went closer to 15 or 20 percent, we were still able to get the overall k-effective down. So we're actually less apprehensive than we were a year ago at least about the fuel storage portion of this.

Another thing I want to talk about is you're going to hear about a lot of reg guides in the next couple of days and probably over the next month or so. We did get a lot of public feedback on Reg Guide 1.240, which is fresh and spent fuel pool criticality analysis. There were a lot of commenters that felt that we should actually, you know, make a revision to this reg guide under the rulemaking. We would like to point that, if you look at that reg guide, as well as any NEI 12-16, which is the document that the reg guide endorses, there's really no specific ties to enrichment levels. There is Section C.1.O, which specifies that the document's recommendations are based on existing fuel applications that are in widespread use and that new and novel configurations and concepts implemented in the future might require additional justification for continued use of the assumptions and recommendations of the reg guide.

Despite the feedback that we got that we should consider updating the reg guide, we did not get any new or novel approaches to review. So we're still going to plan on next updating that reg guide as part of the normal reg guide update process sometime in the future when we receive new and novel approaches from the industry.

Next slide, Aaron. So what the changes to 50.68 are going to look at, there's really only one. We're changing that paragraph (b)(7) and we're changing it from just, you know, 5 percent by weight to an option of either 5 percent by weight or to the value specified in the operating license. We're not making any changes to any other paragraphs, and the reason, I think if you all would recall, last year, we were just opting towards changing it to the value specified in the operating license. We decided not to do that because we determined that there are some BWRs that use an alternative k-infinity criteria for ensuring subcriticality in the spent fuel pool. That's essentially like an alternative to the k-effective and enrichment limit that most other licensees use. So we didn't want to get into a situation where we would, you know, basically take someone out of compliance if they didn't have a value specified, an enrichment value specified in their operating license, which is really the tech spec design features.

Next slide, Aaron. So the overall benefits of this, we think, are that, you know, the k-effective safety limit specified in paragraphs (b)(2), (b)(3), and (b)(4) are maintained at their current levels of 0.95 at 95-percent confidence and probability. Of course, there is the 0.98 for the optimal moderation. But, like, basically, they're kept at the level that they're currently at.

We're going to use the fuel transition LAR process to review any increased enrichments safety impacts that we see coming forward. So, you know, as part of the reviews that we're already performing for fuel transitions, we're going to be, you know, basically making sure that they're adhering to this process, that they have the appropriate limits in their tech spec.

Another advantage of this is that it will allow for any licensee or applicant to utilize whatever enrichments in the HALEU range that they want to utilize. They won't necessarily have to go to a higher value beyond what they would like to analyze, and they basically get what's easiest for them, which is very amenable to the fuel transition LAR process as it reviews new and spent fuel criticality effects.

As I kind of alluded to in the last slide, this alternative, we think, is also very good because it preserves existing 50.68 (b) compliance for all light-water reactors currently operating in the United States. And any increase enrichments LARs would be considered to a voluntary regulatory initiative and, therefore, backfit and forward fit considerations would not apply.

Next slide. Are there any questions or comments for criticality safety? All right. Well, I'm hearing none; so, if that's the case, then I'll hand it off to Jason Piotter so we can keep moving things along. Thank you for your time.

MR. PIOTTER: Thank you, Charlie. Good morning, members, colleagues, and guests. My name is Jason Piotter, and I am the team leader for the new fuels team in the Division of Fuel Management in the Office of Nuclear Material Safety and Safeguards. Prior to that role, I had been a senior technical reviewer in the division for approximately 21 years in structural, thermal, and containment areas. Today, I'd like to briefly discuss our consideration of fissile material package requirements contained in 10 CFR 71.55.

Slide 2, please. So the bottom line up-front is what I'm going to start with today. Transportation packages for UF6, have a certification pathway for enrichments up to 20 weight percent U-235.

While our current regulations, as written, are sufficient to transport higher-enriched UF6, we are proposing a non-mandatory modification of the current enrichment limit that allows for more regulatory certainty while maintaining safety.

If you'll note in this slide, the current regulations reference 71.55(b) and 71.55(g), noting here that the regulation of interest is actually 71.55(b) in the sense that it is the starting point for evaluation of these fissile material packages. And it basically is saying that, when these fissile material packages are evaluated, they need to be evaluated with moderator present. For UF6 packages in particular, an exception was introduced in 2004 that provided a pathway for certification of those packages provided that the enrichment limit of the UF6 is below 5 weight percent or below.

Now, the rest of the regulation for 10 CFR Part 71, in general, do not have an enrichment level limitation for the fissile material. But, again, in this instance, 71.55(g) does have a specification and an exception for UF6 packages. I will note, and I'll get to this a little bit later in the presentation, there is another regulation in 71.55 that does not have an enrichment limit associated with it that allows for not considering moderator inclusion into the package. But, again, we'll get to that in the future slides.

Next slide, please. So absent utilizing the provisions in 71.55(g), applicants for a certificate of compliance have the option of evaluating fissile material transportation packages in a variety of ways. I've already mentioned 71.55(b). They could seek an exemption to 71.55(b) as part of their licensing options, or they could seek an exception, again, to the water and leakage requirements of 71.55(b) using the provisions in 71.55(c). The difference there is that there has to be a special design feature present such that no single failure of packaging would result in an inadvertent criticality in the package.

Next slide, please. So recall from last year for the reg basis we identified three alternative actions that the NRC could take: no rulemaking utilizing the existing certificate of compliance options that could be chosen by industry, rulemaking to increase the enrichment limit to less than 20 weight percent U-235, or rulemaking to remove the enrichment limit in its entirety. The staff initially recommended Alternative 1 simply because the regulations, as written, were then and are still sufficient to certify UF6 packages up to 20 weight percent and the financial analysis itself demonstrated that the options were cost-neutral at the time. In addition, there was not an evident demand signal from industry that there would be a significant number of additional certificates of compliance required, and the NRC had also just recently licensed the DN30-X package which allows for transport of UF6 enriched up to 20 weight percent.

While the staff specifically requested additional information on the need for new COCs for existing packages for new designs, comments from the public did not answer that question directly. The public comments did, however, specifically indicate regulatory certainty as a factor that was important for consideration. Public comments also specifically indicated that an exception for UF6 transport for the enrichment levels in the 5 to 10 weight percent range would also benefit industry. Finally, the public comments indicated the NRC neglected to consider downstream non-rulemaking costs due to the increased number of shipments that would be needed if moderator must be considered as part of certification. The staff looked at these comments and took them into account when evaluating whether or not to update the rulemaking alternatives.

Next slide, please. As a result of additional staff discussion, as well as the public comment, the staff determined that updating rulemaking alternatives was warranted. The updated alternatives include no action; increase the enrichment limit in 71.55(g) to include enrichments up to 10 weight percent; and, three, increase the enrichment limit in 71.55 to include enrichments up to 20 weight percent U-235.

Next slide, please. The staff is now recommending Alternative 2, increase the enrichment limitation for the exception in 71.55(g) for the following reasons: It allows for a modest reduction in regulatory flexibility found in the current regulations in exchange for a modest prescriptive approach for an expanded, yet limited, range of enrichments to maintain safety. A regulatory path still exists for a moderator-inclusion exception for enrichment ranges from 10 to 20 weight percent in the current regulations.

This option balances operational needs of industry, including regulatory certainty, with the increasing safety consequences of an inadvertent criticality due to the presence of moderator. The range of enrichment levels for LWR fuels is not expected to exceed 10 weight percent, and the NRC could not conclude that sufficient justification exists for allowing an exception for UF6 enriched from 10 to 20 weight percent that balances both safety and regulatory certainty.

Next slide, please. This slide illustrates the updated or the proposed updated text that will be added to 71.55(g). Note that, in part four of 71.55(g), it would add the 10 weight percent indicator that we would be going up to that enrichment level. And then in five, which would be a new sub-bullet, a design feature that would need to be incorporated to protect the valve or other fill device for impact for contents with uranium-enriched above 5 weight percent and up to 10 weight percent. And this is consistent with what we've done previously for a separate package that was certified under 71.55(c).

CHAIR BALLINGER: So this is Ron Ballinger. To be clear, this does not require a change in the design of the container itself.

MR. PIOTTER: Correct.

CHAIR BALLINGER: Just the valving and stuff.

MR. PIOTTER: Yes. You could consider it as similar to an impact limiter, I suppose, depending upon what the designs look like. That's correct. And what it would allow for is it would allow for continued use of existing cylinders.

CHAIR BALLINGER: Thank you.

MR. PIOTTER: Yes, sir. And next slide. And, again --

MEMBER ROBERTS: Quick question. This is Tom Roberts. The text of this rule implies that impact is the only failure that could cause inadvertent fill; is that right?

MR. PIOTTER: Yes. If you look at the rest of the regulation, it actually looks at the performance requirements under 71.73, which is the hypothetical accident conditions. So provided that the package and the packaging survive the hypothetical accident conditions, which is really a structural and containment question, then that's, in essence, the issue with excluding moderator. So that's really the focus here is making sure that no moderator gets inside the packaging.

MEMBER ROBERTS: Okay. Thanks. So impact is shorthand for hypothetical accident conditions?

MR. PIOTTER: Correct.

MEMBER ROBERTS: Okay. Thanks.

MR. PIOTTER: Yes. And just to clarify, it's one element of the hypothetical accident conditions that starts the sequence.

MEMBER ROBERTS: It just seems like that's very specific in terms of not really covering performance requirement, which is to have defense-in-depth for the hypothetical accident conditions. So it's okay in terms of writing the rule, but I don’t understand what you mean by the shorthand accident conditions so, as long as it’s clear, you know --

MR. PIOTTER: I think, within the context of the entire 71.55(g) as written, it's evident with the entirety of what's in the rule currently. But thank you.

MEMBER KIRCHNER: Ron, this is Walt Kirchner. May I ask a question?

CHAIR BALLINGER: Oh, sure. Sorry about not keeping track of who's got their hand raised.

MEMBER KIRCHNER: Well, again, 71.55 is general requirements for fissile material packages. It's not LWR-specific, although I guess most of the activity does support the LWR fleet. We have multiple applications looking at using up to 20 percent enrichment in the case of advanced of reactors, so why wouldn't you just go ahead and provide that certainty up to 20 percent industry-wide? I know this is an LWR rulemaking package, but this specific proposal is rather generic and, as Tom was also indicating, I mean, should be looked at in a performance-based manner.

MR. PIOTTER: Yes. Thank you for your question. It's a multitude of factors. I think, one, in terms of the staff considering whether to make this change in an incremental fashion versus considering this thing all at once, we do have the existing regulation 71.55(c) that would cover packages that are enriched 10 weight percent up to 20 weight percent. I think that was sort of the starting point from staff in the decision-making.

I think the second part of the decision-making is that if you look at the increase in consequences as enrichment increases for the packages, the staff did not feel like there was adequate technical basis for inclusion of that enrichment level from 10 to 20 weight percent for these particular packages. And, again, this is an exception. It's an exception to an existing rule, and it's specific to UF6 packages. So the question is is it in the interest of moving forward without that additional technical basis for 10 to 20 weight percent. Is an incremental approach best, and that was the staff determination was that we could make some incremental progress on providing regulatory certainty within the 71.55(g) framework for UF6 packages, but, yet, unless and until we have that additional certainty with respect to 10 or 20 weight percent, from a technical basis perspective, staff was not comfortable moving to that level.

And, again, if we didn't have 71.55(c) in place, I think there would be a stronger consideration for making the exception in 71.55 (g). But because there are multiple pathways and 71.55(c) also is not a prescriptive requirement, it's technology-neutral, if you look at the proposed language that we have for this, it is prescriptive and it's not technology-neutral. If you do have a different type of UF6 package that comes down the pipe that does not have a fill valve similar to what you have for this design, now you're creating a situation where you're assuming what industry is going to do with these packages.

So, again, we're trying to balance that flexibility within the regulation with regulatory certainty with safety. And so that's the point where we decided that this was sort of the, if you want to call it the sweet spot for looking at an incremental improvement, this is where we landed.

MEMBER KIRCHNER: So this is only for existing cylinders, aka packages?

MR. PIOTTER: This particular update would account for existing technologies that have a valve or other fill device, but, essentially, yes. I mean, you would look at it from existing packages. If you're looking at a new design, we're not going to be able to anticipate what those new designs look like, which, again, that's where 71.55(c) would come into play where that's still a reasonable pathway to reach a certificate of compliance. It's just not one that has as much regulatory certainty.

And, again, I think part of the issue here is that we don't know what these future designs are going to look at, so we can't necessarily anticipate what to regulate against if we don't know what's going to be out in the environment. So, again, regulatory flexibility is important in this conversation, which is why we believe that 71.55(c) is an adequate pathway for some of these unknowns that we still have.

MEMBER KIRCHNER: But then if one goes that way, I have it open in front of me, then you apply for an exemption.

MR. PIOTTER: That's correct. I mean, exemption certainly is an opportunity to move forward, as well. The public comment that we received on that front was that, you know, if we want to stay away from regulating by exemption, we have to provide an alternative. And so that was part of the consideration process, as well. Thank you for bringing up the fact that exemptions are still an opportunity.

MEMBER HALNON: This is Greg Halnon. That's not an exemption to implement 71.55(c), though. It stands on its own, right?

MEMBER KIRCHNER: Yes.

CHAIR BALLINGER: And that was Greg Halnon, by the way, for the court reporter.

MEMBER HALNON: Yes. So the question, continuing on with Walt's, is if you implement 51(c) or 55(c) -- I'm sorry -- is there guidance that prevents the bring me a rock syndrome where it stays, you know, you have special design features?

MR. PIOTTER: No. Currently, there's not. And I think that's one of the issues that we would have to address looking at this. But, again, the purpose and the point of 71.55(c) is to provide regulatory flexibility, and so you have an opportunity to provide regulatory certainty either through the regulation or promulgating guidance. You could consider that even for 71.55(g) where we could change the way that the text is written to promulgate guidance; but, again, we were trying to balance all of these different features within the existing rule to allow for the broadest range of flexibility based on what we anticipate will come in for certificates of compliance.

MEMBER HALNON: Okay. Well, I think, as Walt was saying, many of the advanced reactors that we're going to be seeing coming through is going to be implementing 55(c) and you're going to get a potpourri of different ways of doing it. So I'm just thinking that there might be, at least at some point down the road, where some guidance might be warranted. But I understand what you're saying. Thank you.

MR. PIOTTER: Well, let me sort of do an Uno reverse card here. Is the question on fuels or on UF6 specifically? Because increased enriched fuel, clearly, we'll have to either look at 71.55(b) or (c) as the regulatory pathway. I'm just clarifying to make sure that I understand.

MEMBER HALNON: Yes. I'm talking the fuel itself.

MR. PIOTTER: Okay.

MEMBER KIRCHNER: So the implication here is that, from what I'm hearing, is that to go to 20 percent would require different cylinders.

MR. PIOTTER: It would require perhaps a different design. I don't know that it would necessarily require different cylinders. I mean, the issue in particular with ANSI N14.1, that's sort of changed directions now in the sense that it doesn't have the hard limitation of 5 weight percent. The regulatory authority can make some determinations with respect to that, so going to 20 weight percent would be possible. I think the issue that you run into is the number of transportation evolutions that you need to have because you have to consider moderator. And so it's going to reduce the payload capacity on a per-cylinder or per-conveyance basis. And by not having to consider water and leakage, you don't have that restriction on your payload capacity that's due to moderator inclusion alone. You may still have it based on the total enrichment limit; and, again, as you start to get above, I think the number that I received was about 12.5 weight percent, it starts to get away from moderator and starts to really look at just the mass of the material itself. But, again, you are going to take a reduction hit if you have to consider moderator in whatever cylinder or packaging used for UF6.

MEMBER KIRCHNER: But the implications here are then that in 71.55(g) that there's a specific design and it has this valve body, as you reference in sub-section one. So is that a double valve? Is that a double valve? I mean --

MR. PIOTTER: No. It is prescriptive, and it's not technology inclusive to include this particular --

MEMBER KIRCHNER: That's my point.

MR. PIOTTER: Yes. And we're aware of that, and part of the reason was that we did get feedback from the public for existing technologies that there would be a benefit in being able to ship material up to, I think the comment was at 8 weight percent, but we included up to 10 weight percent because that's the range of applicability that we presume, at least for LWRs. That's why the range was chosen, but that was the implication from the comment that we have is that it's for existing technologies. And we recognize that it's not technology inclusive and that it is prescriptive.

But, again, that's that balancing piece

between regulatory certainty, which, again, this is a pathway for that, versus flexibility within the regulation. And I think, in taking in total, when you look at the entirety of 71.55, you have a significant amount of flexibility in that regulation to make some choices as a vendor seeking a certificate of compliance.

MEMBER MARTIN: This is Bob Martin. I appreciate, of course, the legacy that you're working with. I think it probably does support the near-term LWR needs. I think about, you know, what Walt has brought up with, you know, the question of technology inclusivity, and, you know, if this was truly clean from a technology inclusivity, you might have other rules related to the criticality or what have you instead of kind of a state of, like, well, we recognize, you know, a particular package and everything else is kind of the bring us a rock kind of thing. With what could be a bow wave of new reactors and, I mean, I'm not talking about, you know, very soon, you know, a full supply chain we have to worry about to get to the point where we're shipping things over 10 percent or even over 5. But, at some point, do you think you need to revisit the rule to, you know, put other guardrails? Again, related to some of the things you're talking about, you know, and how you consider the problem. Do we need to say, you know, bring up hazards analysis and do a more methodical approach to defining what the packages really need to be to satisfy the rule. I mean, it's really a rule on a package, right, so we might say the enrichment is immaterial, kind of like we just heard on 10.68, right, that there really are -- the problem is the package. It's not necessarily enrichment. But the focus of the rule has been on the package because that's what the community is focused on, but, as we evolve, and, hopefully, this renaissance is going to happen the way we envision it, it seems like we'd be behind the curve if we're not being proactive with a rule that focuses on what the next package has to look like for folks that want to go above 10 and up to 20. I mean, is anyone prodding you on doing that?

I know, obviously, there's a schedule here to get stuff done. Like Ron said, it's incredible that it's only been a year. But we do hear a lot from the advanced reactor community and their chomp at the bit, but they probably haven't thought about this challenge, you know, as much as they're still just thinking about design and operations. I mean, in the background, are you all working on the next iteration of this?

MR. PIOTTER: Yes. I mean, the short answer is, yes, we are. I think there's a lot of different factors here that we're taking into consideration, but this is not sort of a one-and-done activity for us. I think what we're looking at here is there are so many unknowns looking forward into the future that it's very, very difficult, I think, to pick a path to choose and figure out which one is the correct path when we don't know what we're going to see.

I think what this is reminiscent of is really a recognition of the LWR fleet in general because, again, that's what this particular rulemaking is for is for the LWR fleet. That's what we're looking at is looking at the operational history that exists for that and the needs that exist for that for the LWR fleet.

As more of that information, I think, becomes available for the advanced reactor fleet, we're definitely going to have to make new considerations. And I think the new fuels team, in particular, not to make a shameless plug for the new fuels team within DFM and NMSS, but that's one of our goals within that team is to look at sort of that forward-looking or future-looking approach to how we look at the regulatory framework, how we look at our SRPs, and any other associated guidance documents that go along with that; and it's going to have to be a holistic approach. It's not going to just be this one area of consideration. We're going to have to look at the entire range of our regulatory framework as we move forward as we get more information.

The other part of this is, and I mentioned this at the very beginning, is what does that demand signal look like? We do need information. There is a dearth of information both -- and we presented this even at a workshop that we had two weeks ago -- related to TRISO and metal fuels. There's a dearth of information particularly on the back-end of the fuel cycle but even, in some cases, on the front-end of the fuel cycle about what we expect to see.

So as we're able to get more information and integrate that into our process, we'll simply pivot and move toward looking at whether or not additional rulemaking does need to happen. For us, again, this was an incremental step forward in making sure we provided additional flexibility, as well as regulatory certainty, for the LWR fleet, but it's not going to be off the table to make further considerations.

MEMBER MARTIN: It's very important to be proactive. But I will give you a kudo. Last year, and maybe you said explicitly that you were looking at just exemptions for above 5 and, obviously, this is going to be popular with the immediate customers.

MR. PIOTTER: And just for clarification, I mean, exemptions, obviously, would be one way to do that, and 71.55(c), while not having as much regulatory certainty, is another pathway, and that certainly could be something that we consider, as well as any guidance opportunities that we have there, as well, for 71.55(c). I think, again, the challenge there is is, unless we have a design in front of us, developing guidance for that particular part of the rule is very difficult to do. We'd be sort of throwing darts in the air and not really hitting anything. So if we had designs available or at least proposed designs, we would be able to start looking at that even for 71.55(c) to provide, again, more regulatory certainty.

MEMBER MARTIN: Thank you.

MR. PIOTTER: The last slide was just a summary slide. Were there any other questions from the members?

CHAIR BALLINGER: Yes, for clarification on my part. As a practical matter, any advanced reactor that's also in the LWR, this covers it. It's only nonwater-cooled systems.

MR. PIOTTER: I would say that any fuel type that needs UF6 enriched to X level is going to benefit from this up to 10 weight percent. So if UF6 has a feed material, whether it goes into an advanced reactor LWRs is somewhat immaterial because it's a feed material for either or could be a feed material for either.

CHAIR BALLINGER: That's what I thought. Thank you.

MR. PIOTTER: Thank you.

CHAIR BALLINGER: Okay. By our schedule, we're supposed to drag this on until 10:15.

MR. PIOTTER: I was a little concerned about the --

CHAIR BALLINGER: We could probably try to do that, but I'm pretty sure that we don't want to. So I'm wondering whether or not the staff is ready to just keep going or whether we should take a break of some kind?

MR. BENAVIDES: If we could take a minute just to switch over the --

CHAIR BALLINGER: Okay. So we'll just take a few-minute break to switch out people. Thank you.

MR. BENAVIDES: All right. Great. Thank you.

(Whereupon, the above-entitled matter went off the record at 9:21 a.m. and then went back on the record at 9:23 a.m.)

CHAIR BALLINGER: Okay. Go ahead. Thank you.

MR. MESSINA: Thank you. I am Joseph Messina, a reactor systems engineer in the Nuclear Methods and Fuel Analysis Branch in NRR, and I am going to give you a brief overview on how we are addressing fuel dispersal with 10 CFR 50.46(a) rulemaking. And since it's a big effort consisting of many parts throughout the agency of different disciplines, PRA, materials, and reactor systems, I'll provide an overview to kick start before going into some of the technical basis of transition grade size.

Next slide, please. To provide a little bit of background, this part of the rulemaking is driven by fuel fragmentation, relocation, and dispersal, or FFRD. And to provide a little background, at high burnup, experiments have shown that fuel fragment during LOCA, as shown on the figure on the left, and then if the gliding balloons eventually burst due to pressure differential across the cladding, that fragmented fuel can relocate actually into the balloon region and possibly disperse into the RCS. So you'll see the figure on the right are from first LOCA burst tests at Studsvik, and the openings can be large, rather large. And if the fuel fragments finely enough, that fuel can disperse out of that opening. These phenomena are exacerbated at high burnup, and industry is driving towards increasing rod average burnups well past the 62 gigabyte-day rod average limit. That moves methods or license to date, in order to reach a 24-month cycle, so this becomes more of an issue for us at the present.

Next slide, please. From a regulatory perspective, the 50.46 criteria date back to 1974 when FFRD were not known phenomena. The Commission understood that rods would ductility balloon and burst, but they did not know that fuel could possibly disperse due to the fact that burnup was low at the time.

So acceptable approaches to demonstrate compliance with 50.46 and the regulations have ensured that catastrophic failure of the fuel rod structure and most of the fuel bundle configuration are precluded, so we feel that fuel dispersal would be a departure from precedent. And since fuel dispersal is not explicitly addressed within the current regulations, we thought it was important that we develop a regulatory framework to address fuel dispersal. So the draft proposed ruling, which allows for some flexibility regarding fuel dispersal and DG-1434 directly provides guidance for addressing fuel dispersal within the proposed rule.

Next slide, please. To provide a little bit of background on why we are pursuing the path forward that we are, we are planning to risk-inform LOCA in a 50.46(a) style modification of emergency core cooling system requirements as shown in Alternative 2 here, and that would establish a transition break size based on LOCA break frequency and designate the LOCA's above the transition break size as beyond design basis accidents and allow for best-estimate modeling and more realistic assumptions.

Some of the other alternatives that I'll briefly touch on in the next few slides include Alternative 3, which would be to provide a safety demonstration of post-FFRD consequences in the current framework at 95/95 confidence, just as any other LOCA phenomenon is typically done today. And then Alternative 4 would be to provide a generic bounding assessment of dose and risk insights for post-FFRD consequences. And then Alternative 5 would be to use probabilistic fracture mechanics to show that leaks in large pipes will be identified before failure, precluding the need to analyze large-break LOCAs. And this is based, in part, on EPRI's alternative licensing strategy but takes it a step further and, instead of just using this logic to disposition fuel dispersal, it would be used to disposition large-break LOCAs.

Next slide, please. So I'll provide you part of the reason why we chose Alternative 2 was based on public comments, so I'll provide a little overview of that. There was no really unanimous single alternative recommended by industry, though I'll emphasize that any comment represents many of the organizations in industry, so when I mention where their support was, please note that.

And industry recommendations were strongly based on the qualitative schedule impacts that we had in the regulatory basis, but those were quick and crude estimates. After we published the regulatory basis, we further evaluated the schedule impacts for each of the alternatives, and our opinion on the schedule impacts changed and we actually felt that Alternative 2, or 50.46(a) would be the quickest alternative to pursue that was an immediate consideration.

And then going through some of the public comments that the Union of Concerned Scientists and two members of the public did not support any alternative that allowed for fuel dispersal, and then another member of the public recommended waiting until more research and analysis is performed before we make a decision on fuel dispersal.

Next slide, please. So this slide highlights some of the support across industry on alternatives, except for Alternative 2 which I'll discuss in the next slide. So you'll see NEI supported a concurrent pursuit of Alternative 4 and Alternative 5 in the increased enrichment rulemaking schedule. The BWR Owners Group also supported Alternative 4. They stated that they did not see Alternative 5 as a solution due to the fact that leak-before-break has not been approved for any BWRs to date. And then Alternative 5, Westinghouse supported a modified Alternative 5. And I'll state that NEI and Westinghouse, they support Alternative 5 not as it was written but in the way it's written in ALS so not using in probabilistic fracture mechanics to disposition the entire large-break LOCA but to only disposition FFRD. And some alternative approaches mentioned in some of the comments include comments from Framatome and the PWR OG, which suggested using some of the insights that we used in GSI-191.

Next slide, please. On Alternative 2, NEI, BWR Owners Group, and Westinghouse expressed support for Alternative 2 outside of increased enrichment rulemaking combined with 50.46(c) as a separate rulemaking due to the perceived schedule impact or as a backup to what was done in increased enrichment rulemaking. We'll note that aspects of Framatome's response in a 2023 white paper seem to align with parts of Alternative 2.

And then, in the regulatory basis, we asked a question that was responded to by NEI, Westinghouse, and Framatome, and we received a response that stated that, even with a no-dispersal criterion, Alternative 2 was expected to be reasonable with true best-estimate calculations. Now, we're not pursuing a no-dispersal criterion on this rulemaking for LOCAs above the transition break size, but that's how the question was asked and provides some insight into how this rule could be advantageous for the industry.

So this here shows some of the support graphically in one slide, so you'll see where the support was. It just summarizes the previous slides.

Next slide, please.

MEMBER ROBERTS: Joseph, maybe I'll ask a question now. I didn't see anything in the rest of the package that defined what true best estimate means. Is that a term that you expect to disposition later after you have a better definition, or did I miss something?

MR. MESSINA: We provided some words, and I'll talk about that a little more when I go through the rule language and talk about the specific wording of the rule. I can talk about that a little more, but we kept it -- we didn't go into too much detail due to the fact that this is typically done in other beyond design basis accidents, such as ATWS and station blackout.

MEMBER ROBERTS: The complexity of a loss-of-coolant accident analysis is, you know, pretty high and, you know, without doing some sort of sensitivity study or uncertainty study, it seems like you wouldn't know our true best estimate, whatever that term means.

I know there is a discussion in the rulemaking document on sensitivity analyses, so it does appear that there's some, you know, thought that you would need to do just more than question all the numbers and put it into a single calculation and say that's my true best estimate. But I was wondering if you thought that through, or is this all future work to try to work with the applicants to figure out what you would accept to be a true best estimate.

MR. MESSINA: We've done some thought, but we could end up working more on this in the review of different methods and whatnot. And, yes, you know, while we say nominal calculations for these best-estimate calculations, we would need to look at that there's no cliff-edge effects, you know, if something changes drastically, that number is not going to change by 500 degrees and bring you over the limit if you change something very small.

MEMBER ROBERTS: If you took the mean of a best-estimate distribution, you might get a much different answer than just running a calculation with nominal values. Anyway, just trying to understand -- do you get a sense that this is very important, in terms of the industry feedback, to be able to show the true best estimate or do you think that's just kind of a hoke at this point?

MR. MESSINA: I'm not entirely sure. I think that would just be the easiest alternative. And if we consider the factor of time in getting something to high burnups and 24-month cycles, that would probably be the easiest and quickest way to do it. So maybe they do that until they develop more detailed models on dispersal.

MEMBER ROBERTS: The option of either shutting off burst or go through all the analyses you're about to describe on trying to figure out the effects of the dispersal -- okay. I guess I'll wait and see if it comes up again in that context. Thank you.

MEMBER MARTIN: This is Bob Martin. I will add that, you know, on design basis basis, there's plenty of precedent for consideration of uncertainty. Some of it might be explicit with best estimates plus uncertainty. Some of it, even with ATWS and station blackout, maybe it's fairly easy to establish process parameters because they tend to have low uncertainties.

But in consideration of phenomena, you know, those have to be considered because there is no certainty with their characterization, so applicants

-- of course, I'm considering my own experience working in industry, you know, whether we put biases and still kind of call it best estimate or realistic, it's there. So you can't ignore uncertainties, and I know you're going to talk about that a little bit more, so we're jumping on you a little bit soon. But you can't avoid that, and it will be there and maybe you'll get methods because, ultimately, you'll expect the applicant to justify, you know, their decisions. And that will go back to a characterization of uncertainty regardless of whether it's design basis or beyond design basis.

The difference, of course, is what do those analyses support with regard to the design at a beyond design basis, you know, large-break LOCA, you know, with rule change that you're proposing, well, it will mean something a little bit different and it will start looking at what is design basis for tech specs and what have you and things will move around maybe quite a bit. But, nonetheless, because there already is design considerations for beyond design basis accidents and methods have already been established, and you'll just want something consistent. But the terminology without the word uncertainty is probably --

MR. MESSINA: Yes. And we did struggle on exactly how to characterize it in some of the wording of it, but what we really did mean was to match the work done in ATWS space and station blackout space. And there are varying approaches done in those spaces, so, you know --

CHAIR BALLINGER: Walt.

MEMBER KIRCHNER: Thank you, Ron. Yes. Just maybe this is more observation at this point and the questions will come later, but it strikes me that, to first order, a no-burst criterion is, essentially, in my mind, a breakpoint -- poor choice of words -- a demarcation of importance considering methods. Basically, what I'm trying to say is that, up until burst, you have a design basis accident; and, once you burst, especially if you burst a significant amount, I'm not just talking about leakers but actually significant amount of the core reaches a burst by condition, then you're essentially in a severe accident state because the uncertainties at that juncture become very, very large compared to design basis accident analyses using uncertainties, et cetera. That's the method.

The whole burst phenomenon is likely to be very stochastic and not reliably predicted. And then what happens, once you lose geometry and location of the fuel, it propagates huge uncertainty.

So are you, in your thinking, using some burst criterion as a demarcation in how you approach this? Because without burst, you don't have dispersal, so that's a significant juncture. You can deal with fragmentation in intact fuel geometries, but, once you get to burst, then, as I'm suggesting, the uncertainties just are huge.

MR. MESSINA: Yes. And we're not pursuing a no-burst criterion, but that definitely would be considered when we look at a method, you know, the uncertainty in the ability to predict a burst.

MEMBER KIRCHNER: I would suggest the uncertainties are very large and that the process is likely stochastic, not repeated in experiment other than driving the fuel to failure, like in the Studsvik case. So a rulemaking that goes beyond that point raises significant challenges for the staff in evaluating an application, so I'm just observing that that's a demarcation of significance.

MR. MESSINA: James, did you want to weigh in?

MR. CORSON: Yes. So this is James Corson from the staff. So as you say, there is this big demarcation between burst and no burst. While in the rule itself there's no such demarcation, in the guidance that I'll talk about tomorrow, we do have this sort of demarcation. So on the one hand, you know, the sort of easy way, I guess, to satisfy the coolable geometry concerns is to show no burst, but then the guide has considerations that, if you do burst, these are the types of things that you would need to satisfy. So, again, I'll talk a lot more about that tomorrow morning.

MEMBER KIRCHNER: Okay. Thank you. As I said, that was an observation. I'll save my questions. Thank you.

MR. MESSINA: Thanks. So since we are pursuing an updated 50.46(a), I just wanted to touch on some of the history. The initial development started in the early 2000s, and there were several Commission SRMs, as shown on the screen, guiding the staff work on the rule; and there were about a dozen ACRS meetings and three ACRS letters on this rule, as shown here. I included some of the ADAMS Accession Numbers for the letters and the committee transcripts.

This is part of why we had confidence in pursuing Alternative 2 is because it was vetted by the ACRS, so we knew it was technically mature and ready.

Eventually, though, it was rescinded due to lack of industry interest, but there seems to be a renewed interest in this today.

MEMBER PETTI: So, Joe, just a question on the history since almost all of us weren't in that ACRS to make sure I understand it and maybe Dave Rudland wants to answer. That whole discussion and framework was largely based on this expert elicitation of pipe break frequency, et cetera. But, today, we now have xLPR estimates, so, say, maybe a firmer engineering basis to re-look at all of this through that lens. Is that the fair way to characterize --

MR. RUDLAND: Yes. This is Dave Rudland from the staff. Yes, it is. And we'll be talking about that, you know, in a little bit. Rob will give a brief background of what was done back in that time, and then I'll talk about how we use these new tools to try to confirm that.

CHAIR BALLINGER: This is Ron Ballinger. I think the white paper on 1829 has a discussion of both the history and --

MR. MESSINA: Yes. Next slide, please. So this slide summarizes some of the previous slides. We decided to pursue a modernized 50.46(a) in the increased enrichment draft proposed rule due to support in public comments and the fact that it had the smallest impact on the rulemaking schedule and due to the high level of technical maturity and the amount we could leverage the technical basis and work originally performed in the 2010 50.46(a) rulemaking.

And I'll note that this 50.46(a) was and is planned to be a voluntary alternative to 50.46, so we're maintaining much of the original philosophy with some change to some specifics. And I'll note that LOCAs below the transition break size would not be affected by this rule while LOCA above the transition break size would be re-categorized as beyond design basis accidents.

Also in this rule, we are planning to include high-level fuel technology-neutral performance-based ECCS acceptance criteria that are based, in part, on 50.46(c), that rulemaking, but made a little more performance based. And you'll see how we did that this afternoon.

Next slide, please. So how does this address fuel dispersal. So it doesn't directly address fuel dispersal, but it's expected to facilitate safety demonstrations of fuel dispersal because best-estimate modeling and realistic assumptions are expected to significantly reduce or eliminate the potential for fuel dispersal. While this approach does not explicitly address other non-mechanistic approaches to evaluating FFRD as described in other alternatives in the increased enrichment regulatory basis, other licensing pathways exist, for example through the topical report process, and we believe that our performance-based criteria may allow some flexibilities in that space.

Next slide, please. So I briefly mentioned 50.46(c), so I'll give a little bit of a background on that. That was a draft final rule to revise the ECCS acceptance criteria to be performance based and reflect relatively recent research findings on embrittlement of zirconium alloy cladding under LOCA. The NRC LOCA research program running from 1997 to 2016 identified new embrittlement mechanisms shown on the bottom of the screen, hydrogen-enhanced beta layer embrittlement, cladding ID oxidation, and breakaway oxidation, that show that under the current regulations, the 17 percent local oxidation and the 2200 PCT post-quench ductility is not necessarily assured following a postulated LOCA. This was submitted to the Commission in 2016 and underwent several rounds of ACRS review. We are leveraging some of the work done in that rulemaking and in this rulemaking, as well.

And if you go to the next slide, we'll see that the Commission, this year, responded to our rulemaking package, the draft final rulemaking package, returning it to the staff without action and directed the staff to do a few things. First of all was that the staff should apply an appropriate risk-informed regulatory approach to address the research findings of cladding embrittlement. Staff should evaluate that with other associated technical issues being addressed, such as FFRD and risk-informed treatment of LOCAs, including the draft final 50.46(a). And the staff should evaluate whether specific emergency core cooling system criteria, such as cladding temperature, should be codified or instead addressed in regulatory guidance.

So you'll see from the next slide, you'll see that we are addressing a large part of that SRM with this rulemaking by including performance-based broad failure criteria that account for the 50.46(c) research findings in the rule and associated guidance. And so we would use public comments received on those performance-based aspects of this rulemaking to inform any potential future action on the 50.46(c) rulemaking. Since 50.46(a) is a voluntary rule, we will continue to perform the annual ECCS safety assessments to evaluate the impact of the cladding embrittlement research findings for the plants that do not elect to adopt 50.46(a) and address the research findings. We would also evaluate the research findings within the framework of existing regulatory requirements when reviewing submittals that could result in cladding embrittlement impacts, such as power uprates or burnup increases since, for example these phenomena are exacerbated at higher burnup, the cladding embrittlement research findings.

CHAIR BALLINGER: This is Ron Ballinger again. Just to be clear, folks that do not want to do the voluntary rule are still subject to the new research results related to embrittlement for cladding?

MR. MESSINA: Not necessarily. So we're saying if they go to higher burnups, then, yes, we would most likely ask about it and we have asked about it in reviews. But at current burnup, we would probably not.

CHAIR BALLINGER: Ah, okay. So even at current burnup limits, there's the possibility of getting dispersal if you get failure. And so why not ask the applicants that elect to stay with the original to still comply with the new knowledge related to cladding embrittlement in their evaluation of ECCS performance?

MR. MESSINA: So we do our safety assessments, and that provides assurance that, even if we implemented the 50.46(c) criteria, plants would adhere to them. So they are safe to date.

CHAIR BALLINGER: Okay. That's what the conclusion was for 50.46(c). Okay. All right. I'll have to think about it. We have new information and had it for some time, and so, in your evaluation of ECCS performance for any plant, we would think that you would apply that knowledge in your evaluation. Where am I wrong here? Where am I wrong?

MR. MESSINA: Well, we would have it in the back of our mind and decide whether we would need to ask about it or not. But, John, did you want to -- John Lehning in the back.

MR. LEHNING: John Lehning from NRC staff. So we would consider any of that information, and we have considered that, but we would look at it under the existing regulations, like, if it's under 10 CFR 50.46, it might be under the coolable core geometry and could this new information cause us to have a question about that, and we would pursue it as necessary under that regulation. So that's the thing, we just have to work within the existing regulation for the plants that are under that.

CHAIR BALLINGER: So you're not putting blinders on?

MR. LEHNING: Exactly. I mean, we know that information. We're taking that information into account in the reviews that we're doing under the existing regulations for that plant.

CHAIR BALLINGER: Okay. Thank you.

MR. MESSINA: Thanks, John. Next slide, please. So as I said, this is a big rule and a big rulemaking effort, so I'll highlight some of the scope of work that is associated with this rule. You'll hear from Dave Rudland on how we confirmed the original transition break size later today and work we did there. You'll hear from James Corson on the update to the three 50.46(c) reg guides on the screen, and then we also developed three new reg guides. The first one is on the risk-informed evaluation process to support changes made under 50.46(a) that will be discussed later today, the plant-specific applicability of transition break size will be discussed in January, and then addressing the consequences of fuel dispersal we'll be discussing tomorrow.

Next slide, please.

DR. BLEY: Excuse me before you leave that one. Dennis Bley with a question. Up under your first bullet, are you going to talk some about the internal and external expert elicitation in one of these sessions?

MR. RUDLAND: Yes. When I talk about the confirmation of the transition break size, we'll go over the efforts, the recent efforts on the internal and external expert elicitation, yes.

DR. BLEY: Okay. I'd like to see that. I don't think I ever went through that before. Thank you.

MR. MESSINA: Next slide, please. So as you'll see, this rule is a relaxation of the regulatory treatment of LOCAs above the transition break size, but we believe that it maintains adequate protection due to these three bullets: the initiating event frequency for such events are very low and the NRC will ensure that it is low and remains low on a plant-specific basis, the NRC will ensure that risk increases through the rule are minimal and that there are not large increases in overall plant risk, and we'll maintain regulatory control over LOCAs continuing to review ECCS evaluation models and topical reports and plant-specific implementation of those models in those LOCA analyses, as done to date.

Next slide, please. So as I said, this consisted of several different technical expertises: materials, risk, seismic, and reactor systems. So I highlighted some of the main contributors here, and there are others not listed, but I just wanted to give credit where credit is due.

And next slide. And this brings me to the end of my presentation.

MEMBER KIRCHNER: This is Walt Kirchner. Sorry, Ron. I didn't raise my hand. Could you go back two slides, please? So under each of these sub-bullets, could you just enumerate for the record what you anticipate the agency would be doing to -- using your words, the rule relaxes the regulatory treatment above TBS. Could you just enumerate compensatory measures that you would expect the applicant to take or you would inspect such that you would retain this confidence?

MR. MESSINA: Yes. So for the first bullet, you'll hear a lot more in January. There's going to be a draft guide on plant-specific applicability of the transition break size; and, relatively similar, we plan to do inspections on welds for piping greater than the transition break size, as Dave Rudland will talk about later. And then for the changes and the risk increases due to these changes, you'll hear more about that because we specify what we'd expect an applicant to do in the reg guide that will be presented later on, the risk-informed evaluation process. And then we will maintain topical report review and LAR review of LOCA analyses for the --

(Audio interference.)

MEMBER KIRCHNER: Okay. Thank you.

DR. BLEY: This is Dennis Bley again. 50.46(a) just kind of disappeared for a long time. Have there been any additions or changes to the basis since back when that was first issued?

MR. MESSINA: So, partly, yes. So we realized that the basis was developed well over ten years ago, so we didn't just pick up the basis and say, oh, no, it's applicable. We did our work to ensure that that basis remained applicable today, looking at operating experience, delays in probabilistic fracture mechanics and whatnot, and you'll hear more about the work we did to ensure and re-establish that technical basis from Dave Rudland later.

DR. BLEY: Okay. I'm glad to hear that, and I hope you have sizes, you know, what you've picked up since that time.

MR. RUDLAND: And, Dennis, this is Dave Rudland. One of the things we also try to do is we try to take into account and incorporate the thoughts of the industry back in that time frame and some of the concerns they had with the rule and the way the guidance was written at that time and tried to take those and incorporate those also in the updated basis, the updated rule and guidance.

DR. BLEY: Okay. Thanks. And if we get some details on that, that will be helpful.

MEMBER HALNON: This is Greg Halnon. Just a quick question, not a question but I guess what I'll be looking for. I see the rational between the first two bullets on this, but I don't understand how those two comport with -- why does the third bullet have to be there? We leave a lot of risk that's very low in the hands of the licensee to handle. I wouldn't maintain regulatory control over these if the event frequency is very low and we're making sure that it stays low. So just as we go through this process, I'll be listening for that rational.

MEMBER MARTIN: I had a question on -- this is Bob Martin -- adequate protection. Another feature of protection, you know, is the tech specs; and, you know, under a maximum hypothetical accident, totality, you know, look to LOCA as defining the limits on tech specs as we sharpen the pencil, and we, in some ways, have already seen it, just the, you know, just the best estimate LOCA rule change created situations where, you know, non-LOCAs were becoming limiting. Have you all kind of assessed the guidance on the preparation of tech specs to ensure a comprehensive review of events that, you know, will lead to appropriate definitions of tech specs? I mean, we have to revisit that?

MR. MESSINA: A little bit of a hint into some of what we'll discuss later is, you know, we do ensure that anything that is credited for LOCA is placed into tech spec, so we will maintain regulatory control over that. We don't expect there to be major, we didn't think there would be major changes needed to overhaul the tech specs.

MEMBER MARTIN: So that third bullet kind of relates to tech specs; is that what you mean or part of what you mean?

MR. MESSINA: Yes.

CHAIR BALLINGER: With respect to item one, we've had a few precipitating events in the past. V.C. Summer and Davis-Besse and the like, which have resulted in a lot of changes in how we do inspections, where we do inspections, how often we do inspections. And is that applicable to assuring adequate protection? In other words, this is modified in inspection regimes that we have. Does that constitute assurance of adequate protection or could that?

MR. MESSINA: I'll hand it over to Dave Rudland.

MR. RUDLAND: So everything we've learned from the time that this rule was originally written to now has been considered in the development of this basis, and the thought process of the industry's efforts to optimize inspections and how that relates to the relaxation that's happening because this rule was one of the major considerations that we took in developing the rule and the basis.

So I think it's all being considered, and we'll, again, go through that here in the upcoming presentations.

CHAIR BALLINGER: Sprinkled in the various documents which I've read, which I can't remember the exact numbers all the time, that type of wording about the value of inspections and the like is sprinkled, if you want to put it that way, through these documents.

MR. MESSINA: Okay. If no more questions, I'll hand it over to Robert Tregoning.

CHAIR BALLINGER: Okay. Wait a minute. We have sufficiently dragged this on so that, if it's convenient for you folks, this is a good time to take a 15-minute break. Is that good enough? Okay. So let's take a 15-minute recess and come back in a little after 10:15.

(Whereupon, the above-entitled matter went off the record at 10:03 a.m. and then went back on the record at 10:17 a.m.)

CHAIR BALLINGER: Okay. We're back in session, and I think, Joe, are you still on or it's -- no. Okay. Rob.

MR. TREGONING: Yes, Rob Tregoning, senior technical advisor for Research, part of staff. I'm going to come and give a little bit of history. Member Petti said we developed the original technical basis for 50.46(a) about 15 or 20 years ago.

Before I start, and I hate to do this, I don't mean to be openly antagonistic at the beginning of an ACRS meeting, but I feel like I need to correct the record somewhat. Chairman Ballinger indicated that it was a heroic staff effort to do this rule within a year. I just want the record to reflect we started work in March, so it's really been closer to nine months. I think there were many times when the staff would have killed to have gotten another three months to put all this stuff together, so I just wanted to make sure that the record --

CHAIR BALLINGER: And what's the uncertainty been?

(Laughter.)

MR. TREGONING: It's at least 20 percent. So what I'm going to do is just give you a background, and I am going to basically tee Dave up because you'll see that a lot of what we did 15 years ago, we followed a very similar playbook to confirm the continued adequacy of the technical basis over the last nine months or so. So I'm going to give you enough of a flavor to set so that, when Dave presents what he's going to present, you'll have a background and an understanding of the motivation and the approach that we used to try to make sure that the technical basis is still acceptable moving forward.

Next slide, please, Phil. So, again, what I'm going to do, I'm just going to summarize the historical and, when I say historical, it's the pre-

2024 technical basis that we used to develop the TBS, but you'll see a lot of those concepts, approaches, and techniques are very similar. We've just got updated tools compared to what we did in the early 2000s.

There's three components to what I'm going to talk about, and these go chronologically, but also they reflect how the staff's thinking and basis was used to develop the TBS. We first started with assessing LOCA frequencies. And Member Petti indicated that it was by expert elicitation, but it was a blended sort of elicitation. And I think it will be helpful to refresh the members' minds in terms of what was done, so we'll talk about that NUREG-1829 basis. We used that as a starting point. We used that effort as a starting point to select the TBS, but there were other considerations that went into actually picking the TBS, and I'll spend a couple of slides talking about the other considerations that we had.

The one thing that we didn't do in 1829, it was really geared toward looking at normal operating events and anticipated transients, things that a plant would see over their operating life, so sort of expected events. We didn't look at seismic or, in particular, rare seismic events. And we recognized that gap from the very beginning, and we knew we would likely need to do some additional work. So a couple of years in on working on 1829, we started an effort to evaluate seismic integrity of large asset structures and particularly piping and component supports, and that effort was documented in NUREG-1903.

We started that work, we started all this work around 2002 and then both of those NUREGs were published around 2008, so there was a lot of technical basis work that went into underpinning the development of the TBS back in, again, the early 2000s.

Next slide, please, Phil. So I'm going to jump in and talk about 1829: what was the scope, what were the objectives, what we were trying to achieve. Again, the scope is pretty simply stated in that first bullet. We wanted to look at piping and non-piping passive system LOCA frequencies as a function of leak rate. Leak rate is just another way of saying LOCA size. And we also wanted to look at operating effects.

At the time, subsequent license renewal was not on the horizon. We were in the midst of license renewal. I think, at that time, we had maybe half the plants that had been either approved or going through it. So we made sure we did an assessment out to 60 years, but we didn't go, at the time, out to 80 years, again, because it just was something that was not on the horizon. And we used expert elicitation.

So, again, we're focusing on LOCAs that initiate in unisolable portions of the reactor coolant system, and we were specifically focused on LOCAs related to passive component aging but then also tempered by mitigation measures. So, Professor Ballinger, you mentioned Davis-Besse, V.C. Summer. Those things happened really just before we started and embarked on the elicitation, so there was a lot of discussion about not only those events but operating experience in general. I'll give a flavor for how we used operating experience within 1829, and then Dave is going to talk more about in his presentation how we used operating experience since that time to do the updated technical basis work that we've done over the last nine months.

Again, I mentioned this before, but the objective was to determine frequency distributions for typical plant operating cycles and histories. And this third bullet is a key one that we were assuming that there would be no significant changes that would occur in plants' future operating profiles. Why was that? Well, we didn't really have a basis to speculate, so we didn't consider, you know, power uprates certainly had happened but, you know, we didn't consider the effects potentially of whole-scale power uprates and we certainly didn't consider load-following and effects such as that because that's not even something that was discussed back in those times. So there was an expectation that, you know, sort of past performance and how the plants had operated providing steady-state base load would be, roughly, how they would continue to operate into the future.

Next slide. I don't want to spend a whole lot of time on this, but 1829 was groundbreaking in many ways because the LOCA frequencies that we used in PRAs up to that time, they were developed solely based on operating experience and operating history. And there was three seminal efforts that really developed LOCA frequencies, and the first two, WASH-1400 and 1150, they didn't even account for nuclear operating experience by and large. WASH-1400 used oil and gas transmission pipeline failure data to come up with LOCA frequency estimates because, again, in 1975, that was really the best that we had. And then 1150 basically used those same distributions and did a

Bayesian update of the fact that we hadn't had really any LOCAs in nuclear service since that time, so 1150 was really just, again, a Bayesian update of those WASH-1400 estimates.

It wasn't until the mid to late 90s that NUREG/CR-5750 developed specific estimates based on nuclear experience. If you're familiar with 5750, the way it developed its analyses, it looked at precursor events, right, because we haven't had LOCAs but we've had plenty of precursor events. We've had leaks, we've had cracks, we've had degradation due to fact and other things, and it tried to take those precursor events and then extrapolate, given those precursor events, what are my conditional probabilities of ruptures associated with those and then, based on some assumptions, came up with LOCA frequencies that, again, were used for several years in PRA.

When we embarked on 1829, there was a strong recognition within the material and the structural integrity community that operating history by itself may not accurately reflect future performance, especially when you're dealing with very rare unlikely events. And operating experience in and of itself requires significant extrapolation when you're dealing with these larger frequencies that, again, have never happened in service: medium break, large break, and then beyond.

Next slide, please. So very early on, we though expert elicitation was the best way to tackle this, and I'm still convinced that it's the best way, even given the improvements and tools that we've had in the ensuing 15 years. You know, the issue with the operating experience is that LOCA events are rare, as I just detailed on the prior slide. And even with modeling -- and, look, we had PFM models 15 years ago. xLPR is a better model now, but the models are only as good as the mechanism that they can model. Most models are limited in terms of the degradation mechanism systems and loads that they really consider. You can get LOCAs from a variety of sources, and you may or may not have an appropriate model for a LOCA, right. We didn't have a model for Davis-Besse, right, but Davis-Besse was potentially a LOCA that could have happened. In fact, I would argue it's probably the closest we've gotten to a medium-break LOCA over our operating history to date.

And then elicitation has a very strong history and use at the NRC. It was used back in the, you know, 90s in development of the SSHAC hazard curves. In the seismic community, it's been sort of widely used. A lot of the Yucca Mountain stuff, performance assessments for expert elicitation was really prominent there. And then even in the structural community, when we redefined the PTS rule, 50.61(a), that was based on reactor pressure vessel flaw distributions that, at the time were developed via expert elicitation. So we have experience of using this process in rulemaking. I think we've got a hand up, so I don't know if you want to -- Dennis.

DR. BLEY: It's Dennis Bley, Rob. I just wanted to support a little of what you said. Some of the folks I was aware of back then were using that data from leaks, over a range of leaks, to kind of expand it and say, well, you get a larger hole, they just modeled that rowing and completely different mechanisms involved, I think, in looking at the medium and large LOCAs than the mechanisms in the leaks, and I think you guys took account of that.

Are you going to talk particularly about the two elicitation efforts, internal and external?

MR. TREGONING: Yes. I'm going to touch on what we did for 1829, and then Dave is going to elaborate further on what we did over the last nine months.

DR. BLEY: Okay. I'll wait for that to

ask questions about it. Thanks.

MR. TREGONING: All right. Thanks, Dennis. Next slide, please. So this is the approach. I'm not going to go into this into great detail, but I do want to touch on aspects of the approach that Dave is going to follow-up on later, so I want to make sure I tee Dave up.

So similar to what Dave is going to talk about, this internal and external elicitation, we did the same exact process in 1829. We conducted what we called a pilot elicitation, but that was done with internal staff. We used that to help prepare us for doing the formal expert elicitation using a variety of outside stakeholders. So that pilot elicitation was very important, and I'll talk a little bit about that.

Then it was selecting the panel, and the facilitation team was a really important aspect of the elicitation. And then, once we had that team assembled, we collectively worked on developing the technical issues that we were going to assess, and then we developed what are called base cases, and I want to spend a little bit of time on that because Dave is going to talk about how we confirmed or reconfirmed some of those base cases using both operating experience, as well as probabilistic fracture mechanics methods. So it was very similar to what we did 20-some odd years ago. And what the base cases are is we defined, and I'll go into more detail, but we had well-defined scenarios, and then we asked different experts to provide quantitative estimates for the likelihood of such scenarios. And we did it for both piping and non-piping sorts of failures.

So those base cases were very important, and I'll go into those in a little bit more detail. And once we developed those, we formulated the elicitation questions. We conducted individual elicitations, so these were all individual interviews. But then, once we got the individual efforts, we informed the rest of the group on some of the opinions. So even though things were done individually initially, we wanted to make sure that the experts had an understanding of what the other experts had said, and we gave them the opportunity to modify any of their responses to us based on what they had heard from the other experts. I'm not going to talk much about that. In fact, I won't talk anything else about that, but I just wanted to give you a flavor for how it was done.

We spent a long time analyzing the quantitative results and making sure those results matched the qualitative rational that we were provided by the experts because, again, we just didn't ask them for numbers, we asked them to support their numbers with a qualitative basis. In some cases, there was incongruency and, if we found incongruencies, we went back to the experts and said, all right, things aren't quite matching up, which should I believe: your qualitative rational or the numbers that you gave me to support it? And then we were able to rectify things based on that sort of ongoing discussion.

Then we summarized documented results, and then we had internal and external peer review of the process and results. We were done with most of the work for the elicitation, I want to say, around 2004 - 2005. We didn't publish until 2008. That's because a lot of that time was spent doing peer review, uncertainty analysis, sensitivity analysis, a whole range of things that were done, which are documented in 1829.

MEMBER MARTIN: Was there an effort to do a formal failure modes and effects? You talked about, of course, different mechanisms that can contribute to break. I'm kind of curious just how formal the process was. You know, today, of course, methods of hazards analysis incorporate failure modes effects and other techniques. Certainly, those methods have been around for a while, although it might have been pretty new as far as, you know, NRC back in 2004. How formal were those methods to characterize? Obviously, you went to a lot of effort to understand the frequencies.

MR. TREGONING: There was a lot of discussion and effort into modeling and discussing the mechanisms associated with pipe and passive system failure. We didn't do any FMECA analysis on what might happen post failure, right; we were just focusing on --

MEMBER MARTIN: Exactly. On what led to the initiating event --

MR. TREGONING: So the structural community, we don't consider that a FMECA, but we talk about the process or the evolution of damage, right, and aging effects. So those were discussed explicitly, and we actually broke scenarios down into their fundamental attributes or causal factors, things that can lead to potentially degradation and can ultimately challenge structural integrity. I've got a slide that touches on --

MEMBER MARTIN: So it might involve, you know, involving the experts who might make estimates for, you know, hierarchy of failure mechanisms, and then the combination of those leads to a final -- okay.

MR. TREGONING: Look, we had a pretty broad diversity of experts. We had PRA folks. We had structural engineers. We had materials folks. We had plant folks. So each of those people approached this problem from a different way, so we had to have an evaluation framework that would satisfy all those different approaches, and I'll talk a little bit about that.

MEMBER MARTIN: How many people did you have?

MR. TREGONING: We had 12 people. We had 12 experts and then supported by a four-member facilitation team, so the facilitation team were the ones that, you know, worked in conjunction with the experts. And not all of those experts gave us actual quantitative estimates. We got ten PWR quantitative estimates and nine BWR estimates, so some of them actually recused themselves from actually providing estimates and some only provided a subset of estimates that they felt comfortable with.

So it was a pretty good group, though. I mean, it was the most fun I've ever had at the NRC is trying to manage and herd the cats associated with that.

MEMBER MARTIN: Well, you know, one of the criticisms of expert elicitation is that it usually gets dominated by a personality, one or two personalities. Did you find that to be an issue?

(Simultaneous speaking.)

MR. TREGONING: That was not an issue whatsoever.

CHAIR BALLINGER: You're talking about around this table, we're sometimes wrong but never in doubt.

MR. TREGONING: I don't have the names, but most people would recognize, including some that were, you know, future ACRS members. So we had a lot of very opinionated people, and we sought that out. When we picked panelists, we just didn't pick them based on their expertise. That was one consideration. We wanted people that were going to be opinionated. We wanted people that were going to be able to provide a basis for what their opinions were, so that was something that we consciously thought about at the time.

All right. I'm dragging this down. Sorry. Next slide, please. So, again, I'll talk a little bit about the pilot. And, again, I wouldn't talk about this so much, but I want to tee Dave up a little bit.

So the pilot was all done internally. We had 11 internal NRC experts. And, again, we tried to sample a broad knowledge base there, so not only materials folks, structural integrity folks, but, again, PRA, thermal hydraulic folks. And you'll see that we followed a similar template when we did the internal elicitation in this most recent effort.

And we actually got interim results that we used as part of the initial very early-on rulemaking development. We said, okay, here's some estimates that we're seeing, these are some potential break sizes that we might be able to support. So we actually had some results very early on that we could use at least on the side as we were developing the rule at the same time we were developing the technical basis; but, more importantly, we were able to try out and pilot the framework. And so we found things that worked, things that didn't work, and we were able to really tune what we did in the expert elicitation.

And then, as importantly, the internal folks brought up technical issues. They said you really need to make sure that you address this with the outside experts. So we felt like we had a really strong footing to start the external expert elicitation after conducting the pilot elicitation.

I think I've already talked through all these. These were individual elicitations led and monitored by this four-person facilitation team. I mentioned that we had 12 experts, and the facilitation team was largely NRC and NRC contractor subject matter experts.

A little bit about LOCA size classification just because we used some unique definitions at the time, but these definitions sort of served us well. So, historically, LOCA sizes that are used in PRA are based on plant response characteristics, right, if you're doing it under high-pressure or low-pressure injection. So we tried to maintain those consistent definitions, and LOCA categories one, two, and three are really using consistent flow rate definitions and LOCA size classifications that have been historically used in nuclear and certainly NRC regulations, so small break, medium break, large break.

But we really wanted to make sure we went up to flow rates that would be, you know, consistent with the double-ended guillotine break at the largest pipe at the plant. So we defined three more categories for these, I'll call them extra large-break LOCAs that we spanned this large-break LOCA regime from a classical LBLOCA up to even a double-ended guillotine break.

And, again, at least initially, we based on flow-rate thresholds. But then we also developed correlations relating the flow rates to actual break sizes because, again, most of the experts were not thermal hydraulic experts and they felt more comfortably in terms of a break size, so what's a size break that I need to worry about in this particular system. So those correlations were actually incredibly important to make sure the experts could provide opinions and responses based on their subject matter expertise.

We looked at three time periods. The current day, at the time, it was around 2004, and the fleet at that time was about 25 years average operation. But we also did estimates out to the end of the design life, which would be, again, an average 48 years operation and then the end of the first life extension out to 60 years of operation.

Next slide. And you brought up how did we consider, you know, did we consider failure modes and effects. This is how we tried to break things down, right. So you have a general box with your LOCA contributions. You have LOCAs based on passive system and then active system LOCAs. The elicitation did not discuss active system LOCAs at all. Now, this would be something like, you know, valve reliability, pump reliability, pumps running, valves opening when they're supposed to and closing when they're supposed to. And the expectation was those are already handled, we deal with those under the maintenance rule already. We were focusing here on passive system LOCAs.

And then we further broke down passive system LOCAs into piping and non-piping contributions, and then, you know, the piping into the piping systems and then, most importantly, the attributes of those systems. And it's those attributes that provide the causal information that make one particular mechanism more or less likely to be active in that particular system. So the attributes that we covered in 1829 was the geometries of the system, the materials of the systems, the loading history, what types of loads, what mitigation and maintenance is and has been applied in those systems, and then what aging mechanisms are potentially active given these other variables. And those were the different causal factors or attributes that we considered within the elicitation.

The non-piping, we broke them down exactly the same way, but, you know, each component has a different set of attributes or a different weighting of the attributes compared to the pipes. So components that could lead to failures include unisolable LOCA failures, includes steam generator manways, pumps, pressure vessel itself, pressurizer, and valves. But, again, each of those components were also broken down into their attributes. I just don't show it on the flow chart.

MEMBER MARTIN: How does external events and particularly seismic fit into this?

MR. TREGONING: Well, again, as I mentioned at the very beginning, we did not consider seismic. We were only considering normal operating transients, so steady state and sort of anticipated transients, so service at level A and B sorts of loading that you would expect over the lifetime of the plant. That's why we ultimately did 1903 so that we'd get a separate evaluation of the impact of the seismic event. We thought, if we tackled seismic here, we were going to need a whole different level of expertise. We thought it was just going to expand the scope too much.

So we had this structure, and, again, I mentioned we had different experts. Some people gave us very high-level analyses, and some people started at each attribute and worked their way up and developed LOCA contributions at a very fine, very finely-graded area, and we supported both of those approaches. And some experts, again, some experts gave us very high-level information and some, again, as I said, just broke it down in a very detailed manner. And the structure allowed them to provide estimates however they were most comfortable providing estimates.

Next slide. A little bit on the base cases. So we developed base cases, and base cases are important and we use them for anchoring the responses and the base casing conditions specify a unique piping system, size, material, loading, degradation mechanism, and mitigation. And we defined five base cases, two BWR and three PWR base cases. Although BWR Base Case 1 on the recirculation system, there was a 28-inch and a 12-inch, so we really had two separate, we called them Base Case 1a and 1b within the recirculation system. We wanted to get a variety of piping materials systems and then sizes, so you see we have base cases related to the largest size pipes, like the hot leg, the recirculation system. We had more intermediate sizes like feedwater and surge line, and then the HPSI makeup line we wanted to have at least one smaller diameter system. You know, those are typically four to six-inch long lines, and we wanted to make sure that we had a set of base cases that would be more relevant to the smaller pipes because smaller pipes have unique challenges compared to larger pipes.

Now, what we used those base cases for, we actually had a separate subgroup within the expert panel, and they actually did quantitative evaluations of the base case. They calculated each base case as a function of flow rate and operating time and provided LOCA frequencies. We had four separate panel members that were part of this subgroup that analyzed the piping base cases separately. Two of them used primarily classical PRA operating experience types of approaches, and then two of them used probabilistic fracture mechanics using different tools. One guy, Dave Harris, who you certainly well know, used the pcPRAISE code, which, at the time, was probably the state-of-the-art PFM code. And then we had a guy from Rolls-Royce that used the PRODIGAL code, and the PRODIGAL code was the same code that we used in 50.61(a) to come up with flaw distributions. So both of those codes had quite a good lineage and track record at the time, so we thought they were sufficiently mature enough to use in this exercise.

Next slide, please. So I don't want to spend a lot of time on this. This is like a shock-and-wow factor chart because what this shows is the results from the four different estimates that we got from the base cases, so each of those dots represents a different analysis that we got. And then you can see the two BWR base cases to the left and then the three PWR base cases to the right.

I just want to make three points because you can see the scales that I've used. I'm plotting failure frequency as a function of LOCA category, and there's the six or five different LOCA categories that can be represented by those particular base cases. As I think you'll see, there's incredibly large variability, but that was due to the fact that, even though we had well-defined base cases, not every expert could analyze the definition as it was given to them because of limitations in their tools or limitations in the data that they had pertaining to operating experience. So even though we defined the conditions, it didn't mean that everyone could definitively analyze those conditions in a meaningful self-consistent way.

CHAIR BALLINGER: Pardon me. We have a hand held up, but it's not a member, and so I think we need to -- I don't know who this person is. It says Constellation, I think.

MS. SHI: Thank you very much, this is Grace. I don't know if I missed that, can you go back to the previous slide?

CHAIR BALLINGER: Excuse me, and I may be wrong, but are you speaking as a member of the public?

MS. SHI: Probably, yeah.

CHAIR BALLINGER: Okay, so in that case our procedures are that at the end of the presentation we'll have the time for public comment, which we'll take then. If you want to make a comment that's written, you can submit those to the DFO, in this case will most likely be Weidong Wang in the future. So, we apologize for imposing our rules, but those are our rules.

MS. SHI: Is there going to be a section for questions at the end of the presentation?

CHAIR BALLINGER: There will be a session where we will accept public comments. Those comments are not in the form of questions, but if you want to make a comment in the form of a question then you should communicate with Mr. Weidong Wang, the DFO for this meeting.

MS. SHI: Okay, I think mine is more like a question, I don't know if I missed that. Just --

MR. BURKHART: This is Larry Burkhart from the ACRS staff, just to reiterate, this is an ACRS meeting in which the staff and others will provide their input to the committee so they can eventually deliberate on advice to the commission. This is not a public meeting with the staff with a back and forth on questions.

So, there will be an opportunity for a comment later today before we adjourn, and you can provide written comments as Member Ballinger said to myself, lawrence.burkhart@nrc.gov, if you wish. Okay, thank you.

MR. TREGONING: Follow policies and procedures. So, yeah, so even though there's recognized large variability, each base case member went into the entire expert panel and presented what they did, and they took questions, and so everyone made sure that they really understood what the limitations and constraints were associated with each of the different four analyses that were done.

So, and we ask each panel member to critique the approaches and the results from the base cases as part of the elicitation session. So, we got feedback from each expert on how well, or how appropriate they thought the base case analyses were done.

(Simultaneous speaking.)

DR. BLEY: And Rob, just a quick question. I remember what the LOCA categories are, one through six. I see on the BWR case, the BWR-1 results are much higher than BWR-2 across the board. Will you remind me what BWR-1, BWR-2, and BWR-133 are?

MR. TREGONING: Yeah, so BWR-1 was -- thank you for going back, Phil. Sorry for using the shorthand, but BWR-1 was an evaluation of cracking within the main recirculation loop piping. BWR-2 was thermal fatigue in the feed water system, as well as potentially FAC. PWR was SCC in the hot leg, SCC as well as fatigue. Surge line was SCC and fatigue in the surge line.

And then PWR-3 was primarily a fatigue consideration of the HPSI makeup line.

DR. BLEY: Okay, when I look at the curves on the next slide, I probably want to look at all those cases differently because they're different events, but they might combine into an overall frequency within each LOCA category.

MR. TREGONING: Yeah, right, so the different colors are each -- and again, this is a busy slide, I apologize for that, there's a lot of information there that I just haven't explained. But each of the specific colors are representing unique base case. Like I said, each dot represents a different estimate of those base cases.

DR. BLEY: Thank you.

MR. TREGONING: Now, those base cases are very specific events, they don't necessarily equate to LOCA frequencies whatsoever. I'll talk a little bit later about how we use the base cases.

DR. BLEY: Thanks.

CHAIR BALLINGER: If I recall back in those days, and maybe a bit before that, on the BWR side, we were wrestling with BWR pipe cracking, and actually replacing recirculation piping, and all kinds of things. So, that probably has a bearing on estimates.

DR. BLEY: Big time, yeah.

CHAIR BALLINGER: In those, because that was a unique series of events that was going on.

DR. BLEY: By the time we did the elicitation we dealt with IGSCC issues in BWRs, but it had been a long history starting back in the late 80s through even the 90s. And believe me, yes, it was the fact that -- not the repair replacement activities, but the fact that in many cases we'd left cracks in service that we'd mitigated. So, some of the high estimates, in fact the highest estimates were concerned about those sorts of scenarios.

Now, recognizing that when we were doing the PWR results, we were just starting to live through the whole PWSCC issues, because those started in the U.S. in the early 2000s. So, even though PWR mitigation strategies were still evolving, there was certainly a lot of discussion about current operating experience, and what mitigation was likely going to happen.

The experts at the time were probably prescient, because they said we've got this whole basis for how we mitigated IGSCC, the expectation was that they would use many of those same techniques to mitigate PWSCC cracking in PWRs, and that's ultimately what was done. So, again, that's why they were the experts, but it was smart because they knew we had a technical basis, why develop new things if you can rely on a technical basis that largely worked for BWRs?

MEMBER MARTIN: So, a clarification, this is Bob Martin again, you said each plot symbol represents an individual's estimate?

MR. TREGONING: One individual, and we asked them to do a best estimate. Whatever that means, we said give us your best shot.

MEMBER MARTIN: So, basically said you've got input for ten BWRs, and nine PWRs, was that -- I mean, I look at that going --

MR. TREGONING: Well, no, the base cases, there were just four members that did the separate calculations, these were separate calculations. And these were just used to anchor for other people's results. So, these only have a relationship to the LOCA frequencies that they ultimately developed because they were used to anchor other people's estimates.

MEMBER MARTIN: In statistics you kind of consider that a validation set, is that the intent?

MR. TREGONING: Yeah, I didn't want to go too much into this, so --

MEMBER MARTIN: Jump on it now.

MR. TREGONING: When you're trying to

predict very rare events, things that just don't happen. So, providing absolute estimates of rare events is incredibly challenging to do. So, what we did, it's easier to provide relative estimates, I think this is more likely than that. So, we used the base cases to provide a series of quantitative estimates under well defined conditions.

And then the experts said if I'm thinking about recirculation line cracking, and all the different LOCAs that can occur, I'm going to use this BWR-1 as a base case for saying okay, when I think about LOCAs in the recirc system, I'm going to do everything relative to this base case. So, they would say I would think that you're really going to have SCC maybe ten percent as likely as you quantified in the base case.

So, then when you use these relative estimates, and based on the anchoring in terms of whatever base cases they chose to actually develop the quantitative estimates. So, that was really the only purpose of the base cases. Experts can choose to use them or not, many of them did, but not all of them did. So, some of them chose to develop from whole cloth their own absolute estimates.

We had one Westinghouse person use the

SRRA code, which was a relatively recent PFM code, and all of their estimates were developed explicitly from just exercising that code.

CHAIR BALLINGER: Excuse me, we have somebody is making noise in the background, probably need to mute something. But we have two people with raised hands, and number one is Vicki.

MEMBER BIER: Hi, Vicki Bier. I understand your argument about that it's easier to provide relative judgments than absolute judgements, especially for these very small numbers. But one point I want to raise, and I don't think I looked at what you did closely enough to know whether this is an issue.

Anchoring can influence people's judgments a lot even when it's inapplicable. So, for example if you spin a wheel of fortune before you ask somebody for their assessment of something, even though they know it's a random number, and shouldn't bear on their assessment, it still ends up influencing what number they give afterwards, which is kind of not ideal.

So, I don't want to make that as a critique necessarily, because I don't know how your experts used the anchors, but just a concern to be aware of.

MR. TREGONING: Yeah, no, it's a good point, it's something certainly that we were well aware of that at the time. You sort of pick your poison, it's hard to provide absolute numbers, but you do have to be careful when you're anchoring that, because the anchoring in and of itself, like you correctly stated, it can lead to biases.

So, we had elicitation training, where we talked about anchoring, and effects like that to try to make sure that the panelists were aware of it. Do you ever make sure that there are no biases? Of course not. But the one thing I do say, the fact that we had so much variability in those base case estimates, I think was actually a good thing.

Because it provided really a wide range of estimates that people could ultimately use to anchor on based on what they thought was more appropriate or not. And you need to do something, and we thought that again, having relative answers was much easier than trying to just directly provide these very low initiating event -- these very low estimates.

And you'll see that some of the estimates got down to ten to the minus eight per year. So, it's incredibly difficult to even wrap your mind around events that rare.

MEMBER BIER: Okay, I appreciate the comment, thank you.

CHAIR BALLINGER: Dennis?

DR. BLEY: Yeah, I would second what Vicki says, it's bizarre how much that can influence people, even when they know that anchor isn't applicable. Looking at the BWR, I'm guessing that low estimate is all from the same expert increasing mechanistic calculations are coming up with numbers like 10 to the minus 16 and 10 to the minus 12.

I'm not sure what that means to us, but if I were one of the experts looking at this I would probably avoid paying too much attention to that one.

MR. TREGONING: Yeah, and again, these are all the discussions that we had. I will say that you're correct, if you look at those BWR-2 results, and those low ones were all for one particular expert, that expert again, that base case included consideration of FAC as well as fatigue, and his tools were only able to model fatigue failures.

So, fatigue failures in that system according to his estimates were highly unlikely. So, I indicated the notion that we had well defined base cases, but not everyone analyzed the same problem. And understanding what problem they analyzed was part of the discussions that we had with the wider expert panel.

DR. BLEY: Okay, so they understood this sort of stuff going in?

MR. TREGONING: Yeah, again, each of the base case members came in, and had multiple presentations to the rest of the panel, and questions were asked, they each had a write up. The write ups were sort of reviewed, and questions asked, and modifications made, so this was an iterative process as well.

DR. BLEY: Okay, and that was true for both the internal and the external?

MR. TREGONING: No, we didn't do this at all of the internal. The internal is just a pure elicitation. We didn't do any of this base case anchoring, none of that was done internally.

MEMBER MARTIN: I can't help but make the observation that if I was coming in cold, my expectation is that it would follow a power law, right? That's just kind of the rule of thumb for rare events, and for most part you've got to see the power law. And given though you're talking about different kinds of breaks, and I will acknowledge that I'm looking at the document now, and certainly LOCA five to LOCA six, you can kind of see a difference.

But of course it should depart from the power law because you have different things that lead to the break. So, I know you're only giving us a little snapshot here, and I'm the first to throw out the bottom one, because that seems a little off. But everything else seems to follow a nice smooth line, and that would be my expectation going in.

But you would hope that when you started considering all the branches that you showed earlier, that you would see some departure from that. And I guess now that I slide through the document, I do see some of that that's got --

MR. TREGONING: Yeah, you'll see a little bit. Again, when I share the total results, you're coming up, assuming that I get to them, you'll see some of the --

MEMBER MARTIN: No, we're going to keep on asking questions.

MR. TREGONING: That's what we're here for, we were way ahead of schedule, I figured I would torpedo the schedule at this point, but that's okay, that's why we're here. I'm trying to make Dave's presentation after mine much easier, so thank you.

CHAIR BALLINGER: You're trying to force

him to be after lunch.

MR. TREGONING: Well, that's where it is on the agenda.

MEMBER MARTIN: I think Walt had a question.

MEMBER KIRCHNER: Yes, thank you. I just observed that if you were to draw lines through your estimates by your individual experts, these lines are relatively flat, and that's surprises me. What I mean by that is that I would have expected when you got out to LOCA categories four, five, and six, you might see a more distinct drop off in failure frequency.

Could you elaborate? Because you're working up to defining a transition break size, so one thing that would, in a sense from the elicitation, give one confidence to truncate the size LOCA that you're looking at is to see that your estimates for the larger break LOCAs fall off in frequency too, much smaller numbers, but I'm not seeing that.

MR. TREGONING: So, an important clarification, these aren't holistic plant LOCA estimates. These are estimates for various well defined failure scenarios. It makes up part of the LOCA population, but not the entire estimates. So, one of the things that you'll see, especially with the mechanistic analyses, and Dave is going to show the latest results.

Sometimes the mechanistic analysis you can either get a leak, or you can get maybe a larger rupture. So, there's not a lot of demarcation between intermediate size LOCAs, and it's a function of the system itself. And you're seeing some of that here, in some of the results, why they look flatter than you might expect.

Now, when you see the total LOCA estimate curves, they're going to look more power lawish, because again, you have different populations, and you have this cumulative effect where you're summing all these different mechanisms of all these different systems together. So, you're going to get this more average, or sort of aggregated approach, it's going to look more continuously decreasing than you see here in some of these individual scenario estimates, if that makes sense.

MEMBER KIRCHNER: Okay, I'll wait to see the more detailed analysis, thank you.

MR. TREGONING: Well, Dave is going to share some more recent xLPR estimates, and those are flatter than you might anticipate as well, but they actually -- they sort of mimic at least behaviorally sort of the mechanistic effect of frequency as a function of LOCA size that you see in some of these base case estimates. But that's just a teaser, Dave is going to provide that information after lunch probably.

I think we can safely move on. I wouldn't have provided this slide if Dave wasn't going to have a similar slide later, because I knew we would take forever on that slide. We did something similar for non-piping base cases, but it was harder to do. Something I neglected to mention, we also have an operating experience database at the time for piping failures.

A lot of the piping failures, not only in the U.S., but internationally, part of this international database. At the time it was an OPDE piping database, we didn't have anything like that for non-piping. Of course non-piping can consist of things like steam generator tube failures, CRDM tube failures, things like that.

So, one of the things we actually did as part of this, is we actually developed initially a non-piping database that people could use, because something like that wasn't in existence at the time. Now, since then the piping and the non-piping databases have merged, and we've got broader passage system failure databases that have been developed.

But at the time we didn't have anything for non-piping, so we developed this non-piping precursor database. And then we also had prior PFM modeling done of specific issues that we also used for non-piping base cases. And these were past problems that people had worked on prior to the elicitation. For CRDM ejection was something that was very much in vogue.

Because it was something that in the early 2000s, and into 2002, was something that people had developed models on for predicting like North Anna and then what ultimately became the Davis-Besse event. So we had models related to CRDM ejection. And of course BWR and PWR vessel rupture is something that had been studied at that time very extensively, and then even much more so since then.

We were doing 50 61 alpha at the same time, so we had updated PTS evaluation models for PWRs, but we also had events that had been done over the last 10 to 15 years looking at LTOP and normal operating events for BWR vessel rupture, and work that had been done by the industry, so we tried to take advantage of that as well.

And in the analysis we talked a little bit about anchoring, but depending on what scenario the experts were looking at, they had the opportunity if they wanted, to choose what they thought was an appropriate base case. It could be either precursor events, one of the calculated events that we used either for piping or non-piping.

And then they had to provide us a relative likelihood of that particular failure scenario compared to a chosen anchoring, or base case of that. Next slide please. This is just more mechanics, I don't want to spend much time here. But just the big bullets, we developed total BWR and PWR LOCA estimates.

We didn't break things down on a system level, we aggregated each expert opinion all the way up to the highest level, and then we did the aggregation. And that was because we didn't get the same level of consistency, and gradation, and results with individual experts, so we really couldn't combine them until we got up to their total estimates.

Of course aggregation at the time, that was a raging controversy in terms of how we should best aggregate expert opinion. And we looked at a lot of different aggregation schemes, but ultimately what we thought was the best is we thought -- under the premise and philosophy that the group results should be more accurate than any single estimate, we wanted to make sure that the outliers didn't dominate the quantitative estimates.

We did want more of a sort of calculation of the median than again, having outliers drive the final results. We asked experts not to just -- we didn't ask them for means, we asked them for basically quantiles of distribution. So, we asked them your best estimate would be like a median estimate, 50 percent, but it could be higher, or 50 percent lower than that.

And then we asked also for their 5th and 95th percentiles, and we couched them in a way that how much lower could that estimate be, or how much higher could that estimate be. They approached it more from a bounding perspective. Then we interpreted those bounds as more like 5th and 95th percentiles of the individual distributions.

And then we calculated confidence balance associated with each of those parameter estimates, we did a whole boat load of sensitivity analysis, and then the final LOCA distributions that I'm going to show here in a minute reflect both uncertainty, and the uncertainty is quantified in the differences in the individual panel members responses, and in the variability, that was the range of responses among the individual responses.

MEMBER MARTIN: Real quick, how did you avoid outliers not dominating the estimates? I mean seeing something on the earlier slide, 10 to the minus 14, different ways that you can do this if you're fitting a power law you're going to do it differently if you're fitting a linear fit there. Some people, if you had plenty of data you might just drop them. But you don't have a ton of data, so you have to consider --

MR. TREGONING: No, we didn't do -- there was no Olympic scoring, right? We didn't knead out the high and low results, we took all the results.

MEMBER MARTIN: Right, you didn't have enough.

MR. TREGONING: No, but that's to be -- and look, there's a lot of discussion in NUREG about this, but that's why I felt strongly that geometric mean aggregation was the best way to do it, because it weighted all the results, it didn't unfairly weight the outliers. But at the time that was something, because SSHAC, and other elicitations have always arithmetic mean aggregation.

If you use that for very low estimates you're going to be dominated by outliers. And we didn't feel like that that was an accurate representation of what the true panel opinion was. But again, 20 years ago, that was a raging controversy about how we should best do that. You see a lot of discussion about that in NUREG 1829.

MEMBER MARTIN: Thanks.

MR. TREGONING: Next slide please. So, there's the estimates. And I won't go over them in too much detail, but you can see the sort of power law behavior, although you will see some noted drop off. Like from BWRs there's a big drop off with size, and that really reflects when you get to the biggest size, there's just very few components that can result in a LOCA of that size.

So, that's why you get this natural degradation. The PWRs, at least initially you'll see a little flatter curve, and that was really reflective of the time of the expectation that stress corrosion cracking, and events like that, if there was going to be a higher frequency, it would be evidenced in these smaller breaks.

So, that's why it's maybe a little bit

flatter for the Ps, and then it starts to dovetail off. You can see again I've plotted the median, the mean, and the 95th percentile results here. And then the arrow bars reflect the confidence balance. So, those arrow bars reflect the spread in the expert opinions associated with each of those parameters.

And then I've just drawn the ten to the minus five, and ten to the minus six lines just as reference points. Dennis has a question.

CHAIR BALLINGER: Dennis?

DR. BLEY: Yeah, Rob, this isn't a critique of what you guys did, but it's, I think a correction of something you said, two things that you said. Arithmetic mean overemphasizes the high outliers, and geometric mean overemphasizes the lower outliers, so they both have a problem. In the end SSHAC didn't do straight arithmetic mean, they brought the groups back together, and this is the way the methodology tells you to do it.

To come to the group consensus on their best expectation of the view of the technical community. They had some high outliers on some specific things that were really corrupting results. And that really focusing on how to put this stuff together, and what the results are supposed to mean, it gave them a way out of it, and it's a way that makes sense to me a lot.

MR. TREGONING: Yeah, no, thank you for that, Dennis. I was maybe a little bit too glib, and I think your comments are really valid, because you're right, geometric mean can overemphasize the low frequencies as well. We followed, at least conceptually, the SSHAC approach. But we opted not for -- we didn't want to develop consensus estimates.

Now, what we did do, is once we got the initial set of estimates, as well as the qualitative rationale, all that information was provided back to the main group, right? We gave -- so, all of those individual elicitations, it was funneled back to the main group, and then we gave people the opportunity to change their testimony, or the result of the responses that they gave us.

If there was something that they heard that they thought was appropriate to modify their results, I'll say even giving them that opportunity almost no one chose to modify, there were no significant modifications that arose from that, but the discussions that were had were very helpful. Because again, even though things were done individually, we wanted to make sure that at least the group was informed by the rest of the opinions from the other experts.

It wasn't a classic SSHAC approach in that we didn't try to hammer through a consensus approach.

DR. BLEY: And that's generally pretty good unless you have some really strong disagreements, and then --

MR. TREGONING: And that's right, because that's the potential challenge with that. We talked about the fact that strong opinions can dominate. If you try to get a consensus approach, you're more likely to have that sort of outcome, a couple of very passionate individuals will drive the ultimate result. That's why we felt like -- and again, we did mixture distribution, we did geometric mean, we did arithmetic mean.

We just thought based on these particular results that the geometric mean aggregation provided the best estimate of the median opinion of the experts. So that's, I don't mean to give a blanket recommendation of in a vacuum for low frequency estimates you should always use geometric mean, but we thought for this particular exercise, we thought it gave us, again, best expression of what we thought the median results we're being told was from the experts.

But Dennis is giving a sense of the rate of controversy in discussions that we had 20 years ago. It was a lot of fun, and I don't know that we reached any consensus, but here we are 20 years later.

CHAIR BALLINGER: Vicki?

MEMBER BIER: Thank you, this is Vicki Bier again. I'm a little surprised at the low diameters, the uncertainty seems to be quite small. And I don't know if that's because it's being driven by empirical experience in that range, or whether you had assumptions that drove higher uncertainty for larger diameters, or if you can comment on the reasons for that.

MR. TREGONING: Yeah, you have to distinguish between the Ps and the Bs. If you look at the BWR, there's still a fair uncertainty associated with the smallest breaks for BWRs. But for Ps the smallest breaks are dominated by steam generator tube rupture. We've actually good operating experience and data. So, there was an expectation that yes, steam generator tube ruptures for these very small breaks, that's going to be your highest risk driver.

And that's not going away regardless of what you do, because those components have far and away the largest margin of any other passive -- sorry, smallest margin of any other passive component in the plant. So, that's why those particular very small PWR estimates, the uncertainty is much smaller there than for other estimates.

And on the other end when you're trying to estimate rare events, you would expect the uncertainty is going to increase, because you're measuring frequencies that are highly unlikely. So, you should expect to have greater uncertainty associated with those. And that's actually what the results reflect. You did see the uncertainty increase with the threshold break size.

MEMBER BIER: Thanks, that makes sense about the steam generator tubes.

MR. TREGONING: Okay, I think we can move, can we safely move forward? So, summary, and I want to wrap this up because we've got to get through to the other two components. The formal expert elicitation, had these piping and non-piping base cases, panelists provided quantitative estimates supported by qualitative rationale.

And then group results were determined by aggregating the individual panelist estimates, and that's -- I think I will keep going, I want to get into the other meaty parts if there's any more questions. So, I'm going to shift gears now, and talk about how do we use 1829 then to develop the transition break size.

So, the first bullet, we used -- the 1829 results were just used as a starting point. And getting to Dennis' point, we looked at all the uncertainties, we looked at the different ways that you could aggregate the expert opinion. We looked at both mean, and 95th percentile results, and we decided at the end of the day that we're going to use the 95th percentile results so that we make sure that we have some level of conservatism.

And our initial acceptance criteria as we're looking for frequencies less than ten to the minus fifth per year. Depending on uncertainties, given the 1829 results, that puts you in a BWR range of a break of about 13 to 20 inches, and a PWR of about 6 to 10 inches. So, that was our starting point, we said okay, this is an initial range.

But we realized these other things were just as important. So, whatever we pick from the GBS, it should accommodate uncertainties, right? So, we need to make sure we do that. We need to consider at least other types of LOCAs that weren't covered in 1829. Active LOCAs is one, low generated LOCAs from heavy crane drops, severe water hammer, those were things that we didn't explicitly consider, and then seismically induced LOCAs.

Which again, we had a separate exercise related to seismically induced LOCAs. But then this last pool, it was really probably as much of -- was as important a consideration as any of the previous ones. As we looked at the actual populations of pipe sizes that were out in the plants, and what systems they corresponded to, and what operating experience we had associated with those systems.

And that's ultimately, we weighed all of that along with the 1829 results in the final selection of the transition break size. But then the next slide will give the TBS definitions. And there's separate definitions for the PWRs, and BWRs. So, the TBS is defined as a pipe break that's the size of the cross sectional flow area of the largest pipe attached to the main coolant loop.

And for PWRs, that's the size of the largest pipe attached to the cold or hot leg main loop piping, and that's on the order of again, these are plant specific, there's not one number, but it's on the order of 12 inches, maybe anywhere from 10 to 14, maybe even a little bit smaller. And for Bs it's the size of the largest pipe in either the residual heat remover, or feed water systems inside primary containment.

Again, this can vary, but it's somewhere on the order of 20ish inches. And the rationale, the next larger pipe breaks, they're significantly less likely, they're more leak before break resistant. There's a smaller population of those pipe sizes. And then pipe sizes less than the TBS were ones that we typically see experience degradation in terms of operating experience.

And the other reason for picking these sizes as we did, we wanted to make sure we had as much regulatory stability as possible. Because we recognize that passive system failure frequencies can certainly fluctuate over time depending on what's happening. But as we defined the TBS, we wanted to come up with a TBS that we thought would provide us stability.

So that even if these things estimate -- sorry, fluctuate in the future, they're not likely going to require a new TBS definition to maintain acceptable risk. So, those were principal considerations that went into defining the TBS. Next slide.

MEMBER MARTIN: Just real quick, trying to make a connection between 1829. In your categories in 1829, and again, these were two different efforts, I appreciate that, you have the smallest large break at flow rate of greater than 5000 GPM, using that definition you're still in large break land. I mean, LBA, LBB --

MR. TREGONING: Yeah, you're still in large break land, you're beyond -- I mean a lot -- the threshold for a large break LOCA size is plant specific, but a lot of plants it's around six to ten inches maybe, depending on the configuration, and what their ECCS system is designed to handle. But six inches is maybe a rough metric. So, yeah, we're in large break land.

MEMBER MARTIN: When you talk about defining large break, I mean you're still in large break land, you're just kind of shaving off.

MR. TREGONING: We're not -- again, I consider large break land anywhere from six inch up to a double ended guillotine break in the largest pipe in the plant, and that's a very wide range.

MEMBER MARTIN: Thanks.

MR. RUDLAND: I think also for your categories, it's LOCA three category and above, right?

MR. TREGONING: Yeah, we defined LOCA three as the minimum large break, yeah.

MR. RUDLAND: Half of the categories you see are still --

MR. TREGONING: Yeah, because that was the focus, because we knew we were going to be changing sort of the design basis. So, we really wanted -- and that was the hardest thing to predict. So, that's why we had a graded approach to try to look at all these different categories of effectively large break LOCAs. Thanks for that, Dave.

All right, I spent a lot more time on 1829 than I'm going to on 1903, just from a time consideration, as well as a consideration that 1829 was really fundamental to a lot of the other work that you are going to see presented here today. We did have a very extensive effort to look at seismic risk, and I at least wanted to spend a couple of slides talking about it.

So, the purpose of this was to fill in to some of the gaps that we didn't address in 1829, and make sure that seismic risk was going to be acceptable. Now, when we started in 1903, we had a pretty good understanding of what the TBS was going to be. So, we were focusing 1903 on just evaluating breaks that would be bigger than the TBS.

So, we weren't concerned about smaller breaks, or fragility of components that could lead to smaller breaks. We were really just focusing on those bigger breaks that we were looking to move outside of the design basis. And 1903 documents six supporting activities, these are a little bit backwards, but this is how it's written in the reg guide, so you can take issue with the reg guide.

But there was a lot of review of past earthquake experience, especially earthquake experience and how it affected passive systems. There was a review of past seismic PRAs up to sort of the early 2000 time period. And then there was -- and this is dated, but it was actually really good work. In the mid 80s there was a Lawrence Livermore study that looked at both direct and indirect seismic piping rupture.

And it was used to support at the time, GDC4 revision. And we spent a lot of time really combing back over that mid 1980s work, because a lot of it was relevant, and trying to see if we could refresh that in a way to come up with new insights that we could apply at the time to this TBS rule. So, we looked at a lot of past work, and then we did analyses on three particular things.

We looked at the likelihood of unflawed piping failure, flawed piping failure, and then indirect piping failure. Indirect failure is a failure of another component that then leads to a failure of a passive system pipe. So, it could be a failure of component support. Component support fails, and then you get higher loads in the piping system, cause the piping system to fail, it could be a failure of snubber. It could be a failure of a secondary system that causes a missile.

All of these things are considered indirect failures. I'm not going to talk about unflawed piping failures, because 1903 said it's almost impossible to fail unflawed piping systems even under rare seismic events. And there's a lot of basis for that, both in operating experience, as well as experimental work, and analyses.

And Fukushima, which hadn't happened at the time, was another good sort of operating data point which showed that the passive system components, due to both the tsunami, and the seismic event, performed flawlessly. And that's been typically the operating experience that we've seen. It's the pipes themselves are usually quite robust if they're unflawed.

So, we were focusing on things that hadn't been looked at as much, and that's the likelihood of failure if you've got flawed or degraded components, and then indirect piping failure. And we use a mix of --it was largely a deterministic analyses, but we had some probabilistic insights that we applied as well, as part of the analysis.

Next slide please. So, we looked at direct piping failure, and we were focusing on these rare seismic events. And again, I think I said this already, but we were focusing on systems with diameters that were greater than the TBS. So, this would be like the hot leg, cold leg, cross over leg in PWRs.

One point, 1903 really only looks at PWRs, because that was the only set of plants that we really had good design and operating information on. We based a lot of the information in the design that we used on leak before break submittals, and information that had been provided there. So, as I think someone mentioned earlier, there's been no BWR that's been approved for leak before break.

So, we didn't have the same wealth of information to really do a BWR study. However, there's certainly a recognition, and an understanding that even though 1903 was focused on PWRs, a lot of the same insights and results should directly apply to BWRs as well. At the time we used the most recent seismic hazard curves for plants east of the rocky mountains.

Now, since those curves have since been modified significantly as a result of efforts that were ongoing even at that time, but then certainly spurred further by the Fukushima Event, and we looked at developing -- again, we were considering very rare events. So, we were looking at developing seismic stresses for the ten to the minus fifth, and ten to the minus sixth per year seismic event.

And we did that through a very, I'll say robust, and detailed scaling method to take SSE stresses and scale them up to these more rare event earthquake stresses. And even though we used the scaling approach, it was intended to be a best estimate scaling approach, and not overly conservative.

But again, we talked about this earlier, with any sort of scaling approach you are going to have some uncertainties and biases, so I would argue the scaling approach probably skews a little bit conservatively, even though we tried to be best estimate. And then the same thing with indirect piping failures, we tried to analyze, or we did analyze indirect piping failures under rare seismic events.

But we really only looked at a couple of different scenarios, and we were really focusing on large component support failures that may lead to piping failures, and there was a crude assumption that if you had the support failure, then you would ultimately have a conditional rupture frequency of the pipe of one, so a very conservative, crude assumption there.

And we really relied heavily on these prior Lawrence Livermore study results, which we just updated using, at the time was new hazard and ground motion information that wasn't available in the mid 80s, and we developed a mean failure probability of components. Next slide please. So, you can boil down all of the direct piping analyses into this one slide, and there were a number of similar slides.

But this shows we looked at 26 PWRs that we analyzed, so each of these data points, and you can't quite see all of them, but each of those data points is a plant specific evaluation. We looked at a couple of different materials. The yellow curve considers an austenitic pipe and weld, and the purple results are ferritic pipe and weld.

And these look at the likelihood of failure or rupture due to a very long surface flaw. And by very long, a flaw that would be about 80 percent of the entire circumference of the pipe. So, very long surface flaw under a very long seismic event. So, the loading here, these were scaled loadings up to what was expected to be a seismic event with a ten to the minus six per year return frequency.

So, you can see the spread there, the spread there is related to largely the seismic stresses that would be applicable for that particular plant. The main thing that you see is even though this wasn't intended to be a bounding analysis in any way, but what you do see is that for the -- even though we looked -- other than the fact that we used very large flaws.

That even when you consider these very large flaws, you still need a relatively deep critical flaw depth before you get rupture, even under these very seismic loads. So, I'm sorry I neglected -- the scale on the right is the critical flaw size, flaw depth, nominalized by the piping thickness. So, that is the flaw depth at which failure was predicted in the modeling that we did.

So, as long as you are below, or to the left of those curves, you would predict that you would survive the event. If you were on or above the curves, you would predict that you failed that particular event. So, even in the limit, the highest seismic stress is up to about 35 KSI for these very long surface flaws using an austenitic stainless steel weld, you would predict that you would need about a flaw depth of 30 percent through the thickness before you would get a rupture.

Next slide. Now, for indirect piping failures, we only looked at a couple of case studies. So, again, I mentioned we did 26 PWR case studies for the direct piping failure analysis, we only did two plants for the evaluation, it was very limited just because we didn't have a lot of information. And we also recognize that the indirect piping failures are -- I mean all these things are plant specific, but those are incredibly plant specific.

So, we just didn't have the bandwidth, scope, or really direction to do something that was more inclusive, and generic. So, we looked at Calvert Cliffs, which is the CE plant, and then we looked at a Westinghouse plant, those are the only two plants that we did. And I indicated that all we did is we used the conditional failure probabilities that were developed in the Lawrence Livermore study.

And then we just updated those based on the update of the seismic hazard curve analysis associated with those plants. So, you can see I just showed the Calvert Cliff results, and the Calvert Cliff results are also presented in the table, which were extracted from the Lawrence Livermore study. You can see we got a small bump due to the fact that the seismicity increased.

But still, the mean results from the two plants that we looked at were still -- the likelihood of indirect piping failure due to failure of component supports during a seismic event was still a mean result on the order of ten to the minus six. But again, it was a recognition, and you could see there at least in some of the results that you get a fairly good spread for specific plants in terms of what their mean failure frequencies would be expected to be.

Based on again, a plant specific analysis for each of the plants that are listed. Next slide.

So, a brief summary of 1903. The thing that they found, and I mentioned this for unflawed piping, the failure frequency is much, much lower than ten to the minus fifth per year. So, well below the initial acceptance criteria that was used to define the TBS. For flawed pipings, critical flaws, or long, circumferential flaws are generally large, as much as 30 to 40 percent of the wall thickness depending on what type of event you're looking at.

And the expectation was conditional probability of breaks larger than the TBS should also be less than this acceptance criteria that we used as the starting point for developing the TBS. Although having said that, we did not do a generic analysis, or we didn't do a bounding analysis to determine that this was always the case for every plant.

Hence you're going to hear about in the January meeting, some plant specific applicability work that needs to be done to make sure that these particular conclusions hold for each particular plant. And I see Dennis has got his hand up.

DR. BLEY: Yeah, once again. Is the likelihood of a long flaw 80 percent of the way around a pipe factored into this? That seems an incredibly large, long flaw.

MR. TREGONING: Yeah, you're right, that was a predefined assumption as part of the analysis.

We did not estimate and factor in the likelihood of having such a flaw. So, they were purely deterministic from that perspective.

DR. BLEY: Okay.

MEMBER KIRCHNER: I was going to ask the same question as Dennis. What would be the probability of such a flaw being undetected?

MR. TREGONING: Well, we have this discussion internally all the time --

MEMBER KIRCHNER: Based on inspection regimes --

MR. TREGONING: Yeah, we have this discussion all the time. And so, you'll get my opinion here, Dave will probably jump out of his seat because he won't agree with it. But I would say it's on the order of ten to the minus two to ten to the minus three, probably something like that based on operating experience.

But I don't really want to go much beyond that. So, Dave, you want to correct me and say how incredibly high I am?

MR. RUDLAND: This is Dave Rudland, yeah, I think it's extremely low, lower than that. A lot of our work that we've done over the last ten or 15 years on PWSCC, it really kind of demonstrates that these flaws will grow through the thickness and leak well before they get to be 80 percent around the circumference, and 40 percent deep unless they're a small diameter, heavy wall pipe.

In that particular case you may get cracks that are very long around the circumference, and relatively deep. But for the large diameter pipes that we're talking about, I'd have to say -- I'd have to guess one order or two orders of magnitude lower than what Rob estimated.

DR. BLEY: If Rob were correct, we would have seen some of these, right?

MR. TREGONING: We have seen large flaws in large pipes, we have.

DR. BLEY: Approaching that size?

MR. RUDLAND: Maybe not as big as that, but we have seen relatively long, and maybe not as deep flaws. Recent operational experience in the French safety injection system showed very long but shallow flaws. So, the propensity for long flaws is there, but we have not -- in the U.S. especially have not seen very many.

MR. TREGONING: And Dave is right, it is the bigger the pipe, the more unlikely you are to get such flaws, because it's more likely that you'll get a leak before break, but it’s highly dependent on the weld residual stress distribution. And we've looked at a lot of residual stress distributions over the years, so we have a pretty good handle.

But I'm not confident that we've looked at the entire population of weld residual stress distributions that re really out there in the fleet. And then if you have repair welds on top of that, that can really skew what your weld residual stress distribution is. So, we've got a lot of work, and I feel like Dave has a very good basis for his statement.

I'm a little bit more of a cynic than Dave when it comes to the particular likelihood of these flaws, given that I recognize that there's a population of weld residual stresses in repair welds that we just have epistemic uncertainty as to what that true distribution looks like.

DR. BLEY: Well, Dave suggested it would be higher for thick walled, small diameter pipe, where do we have pipe like that?

MR. RUDLAND: We've done numerical analyses that suggest that depending again, like Rob said, on the residual stress, that you can get growth around the circumference relatively quickly, because a lot of these residual stress behaviors are kind of sinusoidal, where they go into compression on the back half of the wall, right?

So, it slows it down as it's growing through the wall, but continues to grow relatively fast on the inside surface.

DR. BLEY: Okay, thanks.

CHAIR BALLINGER: This analysis, and when we get to Dave's -- the white paper, all of these large diameter pipes in the U.S., with the exception of maybe two have been mitigated.

MR. RUDLAND: It's a similar amount of welds.

CHAIR BALLINGER: Yeah, the similar amount of welds have all been mitigated. So, the failure probability of those by, I think your own words, is zero.

MR. RUDLAND: Yeah, it goes down extremely --

CHAIR BALLINGER: It goes down extremely --

MR. RUDLAND: I would never use zero when I talk about a failure probability.

CHAIR BALLINGER: Okay, but there are several --

MR. RUDLAND: There are several places in the plants where they're not eradicated, they're managing it by inspection.

CHAIR BALLINGER: So, I'll ask the same question with the white paper, you said the elicitation still holds in spite of the fact that we've done a lot by way of mitigation, inspections, and everything. So, I was a little surprised --

MR. RUDLAND: Just wait until my presentation.

CHAIR BALLINGER: Yeah.

MR. TREGONING: But recognize again, all these things were happening, Ron, during the elicitation, and we did not only current day estimates, but future estimates. So, the future estimates, all the experts agreed that we would have that issue mitigated by the time we got out to 40 years, if not before that. So, in some ways that was built into the responses. There was an expectation that those were going to be mitigated.

CHAIR BALLINGER: And yet they still found the same numbers.

MR. TREGONING: There is a trend versus operating experience, not as much as you might think.

CHAIR BALLINGER: Thanks.

MR. TREGONING: But again, you've got competing trends here. So, the expectation was PWSCC is going to decrease over the next 15 years, but you've still got aging that's going on. So, there's other mechanisms at play here other than just PWSCC and the welds that you've mitigated. Like Dave said, you've got other welds in other systems that you haven't mitigated.

MR. RUDLAND: And degradation that we just don't know of quite yet.

MR. TREGONING: Correct.

CHAIR BALLINGER: Life is 100 percent fatal.

MR. TREGONING: Okay, I think I'm --

CHAIR BALLINGER: One more slide, maybe?

MR. TREGONING: Yes.

MEMBER BIER: I have one question before we move on, Vicki Bier again. It's not really pertaining to this particular slide, but just in general. You mentioned you wanted to have results that would be kind of relatively stable to maximize regulatory certainty. And I understand not wanting to have things that would change by plant status, or minor fluctuations, or whatever.

But I assume that this would all be

revisited if there was some new phenomenon we found out about, or some averse experience somewhere in the world that caused us to rethink this.

MR. TREGONING: Yeah, that's a great question, Dave is going to talk about that a little bit more, and then we'll -- we'll talk about it, yes.

MR. RUDLAND: So, there was some commission direction at the time in 2008, that at the time the commission had asked the staff to reevaluate this on a ten year basis because of evolving technologies, because of evolving mitigations, and things like that, to continually look at that as the plants age.

MR. TREGONING: So, yes. So, there is -- the staff is planning to do this exercise periodically, and especially if it's triggered by some sort of operating experience event.

MEMBER BIER: Okay, thank you.

MR. TREGONING: All right, I think I have one more slide. So, just in summary, we developed passive system LOCA frequencies for Bs and PWR plants using 1829, using an expert elicitation process. And then TBS was increased to address additional factors, and to promote regulatory stability, and then we performed this confirmatory study that determined if the risks of LOCA greater than the TBS due to rare seismic events were acceptable.

Which, 1903 certainly found out, certainly confirmed those events. So, that was the basis again, up to 20 some odd years ago that was again, reported ultimately in 2008 with the release of 1809 and 1903 reports. Dave is going to talk about what we've -- and I'm keying Dave up for after lunch, I'm assuming we're going to break and go with Dave after lunch.

But he is going to talk about what we did to reaffirm the continued applicability and viability of that technical basis work. Because a lot has happened in the ensuing 15 years or so since that work has been published. So, we recognized when we got this challenge, we couldn't just take off, and blow off the dust on this technical basis without evaluating if things had changed either favorably or negatively.

And how we needed to account for those changes both within the rule itself, as well as in the guidance associated with the rule. So, we're not going to talk about the guidance so much today, we're going to talk about that in January. But Dave will talk about some aspects that we used to factor in to changes in the 50 46A rule compared to what the rule looked like in 2010 time frame.

Although recognize that 2010 rule is the basis, so there's a lot of similarities between the current rule and what was originally developed in 2010. And that was all I had, I don't know if there's other questions.

CHAIR BALLINGER: Questions among the members and consultants? Okay, well you've got points for finishing this whole thing in seven months, but you get negative points for finishing 15 minutes early.

MR. TREGONING: That's negative points?

CHAIR BALLINGER: Yeah. We have to go until lunch. So, we're at the right point to take a break, and so I would propose that we recess for lunch until 1:00 p.m.

(Whereupon, the above-entitled matter went off the record at 11:46 a.m. and resumed at 1:00 p.m.)

CHAIR BALLINGER: Okay, it's 1:00 o'clock, we're back in session, and I think Dave Rudland is on for the afternoon, or for however long it takes.

MR. RUDLAND: Okay, thanks, Ron. Yeah, my name is Dave Rudland, I am a senior technical advisor for materials in the Division of New and Renewed Licenses at NRR, and I'm going to talk about the work that was done recently to look at the continued adequacy of transition break size. As Rob Tregoning talked about before lunch, the original effort to develop the basis for the transition break size was very comprehensive, very complex, took three plus years to do.

And when we decided to move forward with an alternative to address FFRD, one of the reasons why we chose this path was to try to leverage as much work as we could that was done earlier. The hope was to be able to take the basis, and just use that, and write the rule that way. But we knew it had been almost 20 years since the release of that proposed rule back in the 2008 time frame.

And that we needed to take a look at whether or not the basis was still valid. We've talked several times about the time frames that we were under. So, we didn't have an exorbitant amount of time to really go into the nitty gritty details, and confirm every little thing about the basis that rob talked about earlier.

But we wanted to take as comprehensive of approach as we could. Next slide. And so, we looked at both the 1829 LOCA frequencies, and the NUREG 1903 results, and we wanted to confirm both of those, and then take a look at whether or not the thoughts that we had, or the changes that we thought might be would impact the transition break size.

So, we took both a qualitative and quantitative look at the basis. Qualitatively we conducted a very similar, but much smaller scoped internal and external elicitation, and I'll go over these details in a few slides. We then looked at the impact of recent operational experience, and this is experience not just from our fleet, but also from the international fleet, and lessons learned from the internal and external elicitation.

We then also looked at some quantitative views. We did a probabilistic fracture mechanic study, not again to develop LOCA frequencies, but to try to confirm some of the frequency estimates that were made in NUREG 1829. We then also did an international operation database study where we took, and made a Bayesian update to the LOCA frequencies based on the operational experience that occurred between the 2004 time frame and the 2024 time frame.

All of this information is documented in a white paper, and the ADAMS session number is there for that white paper. I should point out that in addition to that white paper, there are two technical reports that accompany that. One on the details of the probabilistic fracture mechanic study, and the operational database study, and one on the internal and external elicitation.

CHAIR BALLINGER: Do we have those?

MR. RUDLAND: They are referenced in the white paper. They're publicly available, and in the white paper. Next slide. All right, I'm going to start with the internal and external elicitation. As Rob pointed out, 1829 was based on a formal expert elicitation. They had done a pilot elicitation with the internal staff, and then they had an external elicitation that was based on lessons learned from the internal pilot, and consisted of a dozen or so external experts with a wide variety of technical expertise.

The process that we chose for this mimics that process to really evaluate the completeness, and continued viability of both the 1829 and the 1903 results. Our goal in the internal and external elicitation was not to develop LOCA frequencies, again, but to identify any scenarios that maybe were not considered, that were new, novel, or that were underestimated in NUREG 1829 and 1903 efforts in the early 2000s.

And also it was to look at the likelihood and or technical rule making gaps associated with each of those scenarios. Okay, next slide, thanks. So, this is the approach, and again, it follows the approach that Rob talked about earlier. We had to first select an appropriate panelist internally, and again, our elicitation in this effort focused more on staff, and less on the external.

Mainly because again, of the time frame that we had, it would have been very difficult for us to put contracts in place to get the expert people to participate in the time frame that we had. So, we focused a lot on the internal experts within the agency. We had 13 senior staff, and again, we chose those staff members with collective expertise across all of the relevant areas.

Externally we actually had two of the original panelists that we were able to question that had complementary expertise pertaining to these types of systems. So, that's how our breakdown was in this particular case. Then we formulated an initial set of questions and topics focusing mainly on information and data, and experience since the early to mid 2000s.

Again, as with the original elicitation, they wanted to consider direct, indirect, and common cause failure scenarios, and identify any important quality factors. In both cases, the internal and external, we had all the kick off meetings where we talked about the objectives, backgrounds motivations, discussed and clarified topics, questions, and identifying any initial considerations.

We then allowed the panelists to go off on their own and develop independent responses. Again, we weren't looking for, in this particular case, any kind of quantitative numbers, just more scenarios where the assumptions that we made in 1829 or 1903 may have been impacted, or anything that may impact the failure frequencies for those pipes that are greater than the TBS.

After they submitted their responses, we had follow on meetings to talk about the individual responses, and determine path forwards for each of the open issues. I believe there's a hand up, I couldn't tell who.

MEMBER KIRCHNER: Dave, this is Walt Kirchner, good afternoon.

MR. RUDLAND: Hi.

MEMBER KIRCHNER: How many of your 13 senior staff were what I would call repeats from your original panels?

MR. RUDLAND: I don't even know, was there one? Yeah, so Rob says there was one.

MEMBER KIRCHNER: Good, okay, thank you.

MR. RUDLAND: Maybe two, yeah. So, again, there's not a lot of folks that were here at the agency that were there in 2008, or even before that. I think Rob was here. In fact I joined the agency in 2008, but I was working prior for a contractor who was one of the original panelists. So, I was heavily involved in his input, but not directly involved in the development of 1829.

So, nut much, but the two external panelists were two of the original panelists from 1829. Okay, next slide. So, from the internal elicitation response, there were actually 21 scenarios identified, and those scenarios were a range of different issues. We've binned these here in different categories.

The first bin is those that were addressed through the applicability studies, and that is how a plant specific issue that may need to be addressed to demonstrate that they're applicable to use this transition break size. Some of those topics included embrittlement, SCC, and as such. We had a group of scenarios that we felt were addressed through the rule making itself, the proposed rule that we wrote.

Motivation for future plant changes, PRA, margins, uncertainties, things like that we were able to address within the rule making itself. And then those things that are addressed within the current regulations, and some of that was through all that frequency, water chemistry, and degraded grid stability, and things like that that are already handled.

I'm going to be talking a little bit more about the first two topics later in the presentation. To give an example -- next slide, I'm sorry. To give some examples of some of the topics, I've shown two here, the first is treatment of leak before break piping. And the concern, and the issue was are you giving consideration for plants that have LBB approvals over those that don't have LBB approval.

And through our discussions, the disposition for that was that really no explicit special treatment was given to those plants in the rule itself, but we felt that any individual licensee may be able to leverage their LBB approval as part of the plant specific applicability. And again, a lot of that particular thing, and how they would do that will be in the regulator guide for plant specific applicability, which we'll discuss in January.

The other is the potential for degrading grid stability, and this is would you have a higher risk if the loss of off site power is now evaluated with the LOCA analyses. And if you move the LOCA analyses to be an on site basis that may be the case, and the staff felt that the loop frequencies really don't indicate an increasing trend in those events.

PRA still needs to consider the risk associated with that, and many plants currently employ load monitoring software to predict that type of unavailability. All of these different scenarios, or dispositions in a similar way, and that's all documented in our elicitation report summarized in the white paper I talked about.

Okay, let's go to the next slide. Okay, so, just to provide a quick summary of what the external elicitation was, we asked the two panel members what their thoughts were on frequency of breaks that were less than the TBS, and frequency of breaks that were greater than the TBS. For those that were less than the TBS, kind of had a split view.

One panelist thought that the frequency of breaks are representative that are lower than the TBS, and one panelist thought they were still conservative.

For the breaks greater than the TBS, both panelists agreed that the TBS is conservative, or the frequency of breaks is conservative relative to the TBS. And those opinions were based on mitigation practices, and increased knowledge as we've changed the procedures and the rules from the early 2000s to the 2024 time frame.

So, they really felt that the historical TBS remains viable, and this is the value that Rob talked about earlier. We did ask them again, what possible scenarios may lead to breaks greater than the TBS, and one panelist felt there was really, at this point, no credible scenario where that may happen. The other panelist was concerned a bit about some aging effects, especially those that affect the fracture toughness, like thermal aging, or residual stress effects that are difficult to quantify.

They were also concerned again, about long, shallow surface flows in these low toughness materials, and a propensity for causing a rupture. But the panelists agreed that the likelihood of those is strongly plant specific. I think Dennis.

DR. BLEY: Yeah, David, it's Dennis Bley. On that first one, the frequency breaks less than the TBS are representative or conservative, were your two panelists really far apart on this, or they're about right in maybe they're a little conservative?

MR. RUDLAND: I think they were relatively close. I think where the biggest differences were was on whether or not there would be anything that could possibly -- any credible scenario that could possibly cause a break I those bigger than the TBS, right? But this is frequency of a break less than TBS, I think they were pretty close on those results.

DR. BLEY: Okay, thanks. And just a quick question --

MR. RUDLAND: I think Rob wants to make a comment.

MR. TREGONING: Yeah, Dennis, just to be clear, we asked them to compare their results from 20 years ago with if they had to give similar results, would those results increase, decrease, or remain the same. So, when Dave talks about them being representative of conservative, they're pertaining to their own individual results that they brought almost 20 years ago.

DR. BLEY: Okay, thanks. And did you use the same process this time around of sharing information and seeing if they wanted to change their minds?

MR. TREGONING: Yeah, everything was accelerated this time, so we did everything collectively.

DR. BLEY: Okay.

MR. TREGONING: These are all collective discussions.

MR. RUDLAND: Again, our time frames did not allow a lot of the details that occurred in the original elicitation. Okay, let's move to the next slide. So, some of the dispositions that we talked about, of course, was continued research and monitoring. We've been spending a lot of time talking about the thermal aging for austenitic welds, and cast materials, and I'll talk about that again in a little bit.

The group agreed that harvesting is important to be able to continue to look at those things, to extend aging studies, to look at particular properties at the end of the license renewal period, and to continue to monitor the U.S. international operational experience, and not just for SCC cracking, but for any type of active degradation that could lead to these long surface cracks, especially for large diameter piping.

We thought it was very important even that the output of the external elicitation to make sure that we have plant specific applicability demonstration for the TBS. So, as Joey will talk about later, you'll see in the rule it requires that they have to demonstrate plant specific effects, do not invalidate the TBS before implementing the rule.

And like I mentioned just a little bit ago, there is additional guidance that we'll talk about in January in DG-1428, it provides methods to demonstrate plant specific applicability, and gives options for the plants to be able to find different ways to demonstrate plant specific applicability. Okay, let's go to the next slide.

All right, so from the expert elicitation, we wanted to take another qualitative look at some of the more recent operational experience, and what we did in terms of selecting what we were going to look at was really based on our staff's experience with the operating experience that happened in the U.S.

The input that we got from the elicitation, both the internal elicitation and the external elicitation about operational experience and things to consider, as well as our knowledge of the international events that had occurred, and the disposition of those. And so, we chose these categories, and I'll go through each one of these in detail about what we looked at, and how we dispositioned that, and if there were any impacts that we thought needed to be taken care of as part of this rule.

Okay, let's go to the next slide. The first few are focused on piping, and so the first one is thermal embrittlement, I talked about it a little bit already. The issue really is a decrease in fracture toughness, not only for cast, but also for austenitic stainless steel welds that could impact the failure frequencies.

The staff looked at a variety of experimental testing that was done, developing of aging management programs, which was not, at the time that the original 5046A rule was written, there were not many plants that had applied for license renewal, so the plants didn't have aging management programs. So, we know of the aging management programs.

These particular materials have lack of active degradation, and there's ongoing inspections that occur. So, staff felt qualitatively there's really no safety concern with thermal embrittlement. Anything that would occur would be found through the normal inspection programs. Stress corrosion cracking of stainless steels in PWRs, I think we're all aware that there had been cracks that were identified in safety injection, and RHR lines, and the stainless steel welds in the French fleet, the French PWRs.

That was kind of an unexpected stress corrosion cracking, and it could occur in the U.S., and may impact the failure frequencies. I think we've talked to the ACRS about this effort in the past, but the staff conducted a risk informed analysis using the LIC-504 process to go through those, and calculate whether or not there was a safety impact.

The industry also did a very parallel effort on that, and so the staff, between those two, felt that there was a reasonable sureness of integrity, that if those flaws did exist in the piping in the U.S., there would be no safety impacts. A lot of the stuff, especially these ones that I say have LIC-504s attached with them are all in the public domain also, and are referenced within this white paper that I talked about earlier. Okay, let's go to the next one.

MEMBER KIRCHNER: Dave?

MR. RUDLAND: I'm sorry, yes.

MEMBER KIRCHNER: This is Walt Kirchner again, sorry to interrupt you. Could you go back to the last slide? And give us -- it's nice to know that you did a risk informed analysis, but could you give us the physical reasons why what happened in the French fleet is not expected to happen in the U.S. fleet, based on physics?

MR. RUDLAND: Okay, so the cause of the French cracking is still under question. They -- even if it has not released their root cause analysis. However, staff feel that it's probably two different reasons why it occurred. The French plants were modified Westinghouse designs. So, they took the Westinghouse design, and Framatome modified for the dose requirements and such in the French regulations.

And in that they changed these systems, and made the pipe different such that they had much higher thermal stratification loading at the locations where the cracks occurred. Those particular loads aren't typical, or at all in the U.S. plants at similar locations. So, that's one of the ideas why that cracking occurred, and it wouldn't have occurred in the U.S.

The second is the residual stress, a lot of the welds that we looked at that were sectioned had very large root welds, and because of that you end up with a very high tensile residual stress on the ID surface, and most of the cracks that occurred were very long, but very shallow. And that was mainly driven by the fact that the residual stressors were all in that main root B weld basically.

So, they didn't grow beyond the root B. And it was a very unusual welding practice, typically you don't have those kind of large weld beads, because you need a lot of heat input to do that, and you kind of of -- there's too many weld qualification issues to cause you to have those types of welds. So, it's expected that many of those types of welds would not occur in the U.S. fleet.

In addition, we have done -- we went back and looked through all of the past inspection records for these welds in the U.S., and the industry has implemented a change to the inspection procedure for these welds, in that they now require that they look for SCC, because originally these welds were only susceptible to thermal fatigue, so they were only looking for fatigue cracks, and the MDE is different.

So, they changed the requirement that they would have to use an SCC qualified process to look at these, and for those that have so far, there has been no indications found. Does that help?

MEMBER KIRCHNER: Yes, thank you very

much, that's much better than saying it was risk informed.

MR. RUDLAND: Yeah, we went through all of that in that risk informed analysis. I'll just point out that gain, in that case we leveraged probabilistic fracture mechanics also to try and make an estimate of what the initiating event frequency would be for a LOCA of that size that we then rolled into the PRA, that's why we called it risk informed.

MEMBER KIRCHNER: Well, since you offered that, then it begs the question, did you use the probabilistic fracture mechanics, and estimate how the French designs would perform?

MR. RUDLAND: We did not, no, we did not.

MEMBER KIRCHNER: Okay, so the probabilistic fracture mechanics analysis was just to give you some level of confidence in addition to the other physical design considerations.

MR. RUDLAND: Yeah, we were concerned for the U.S. fleet, so we only did the analyses relative to the designs and the loads of the U.S. fleet.

CHAIR BALLINGER: This is Ron Ballinger, am I to understand that the French site also did a PFM analysis?

MR. RUDLAND: I'm not sure if the French

did. I don't think the French -- the French aren't really -- they're not proponents of probabilistic analysis, so I know they did a lot of deterministic work. EPRI also did safety --

CHAIR BALLINGER: I thought they had been encouraged by that committee.

MR. RUDLAND: Yeah, EPRI also did a safety analysis for this also. While they didn't use probabilistic fracture mechanics, I don't believe, they did leverage risk insights in making their determination of safety.

MEMBER KIRCHNER: Ron, this is Walt again. I might suggest for the future that if you have good data from the French experience, that this would be a good benchmark to try our probabilistic fracture mechanics against. It would be interesting to see what the results show.

MR. RUDLAND: Yeah, the French did a pretty good job of documenting these flaws. So, they have a good little database of what flaws were in these pipes, what the distribution of lengths and depths were, which would be extremely valuable I think for running probabilistic fracture mechanics.

CHAIR BALLINGER: I think we have some of that.

MR. RUDLAND: We have all of that, yes, we have all of that information. Okay, if there's nothing else, I'll move on to the next slide. Okay, the next issue was inspection frequency changes. We've talked about this a little bit, but there are ongoing efforts within ASME to optimize inspection frequencies in developing less inspections for piping and components.

In most cases for piping itself, risk informed ISI is in place at all of the plants. So, they have a focused inspection program on those higher risk locations, but categories may change, the inspection frequencies as more information is given, and as the ASME decides to move forward. And so the staff really feel that continuing inspections on these welds, especially those welds that are in piping greater than the TBS is essential to supporting the basis for the transition break size.

I'll talk a little bit more about what that means here in just a second. So, for all the piping issues that I've talked about that the staff considered, many of the issues were analyzed through our LIC-504 process. Some were analyzed through research or licensing action, or through the industry efforts, and so we really don't believe there is much impact on the piping frequencies from this.

But the staff really does -- did really recommend that for this piping, where diameter is greater than the TBS, that a ten percent sample of the welds greater than TBS is needed for each interval.

CHAIR BALLINGER: But again, all of these pipes, with maybe according to Craig one or two exceptions, have been mitigated.

MR. RUDLAND: No, we're talking about the pipe, the ones that are not mitigated. Yeah, we're talking about stainless steel piping welds that are not mitigated. The welds that we're talking about currently are in a category of risk informed inspection where they don't get much scrutiny, much inspections, because they have no degradation, and they have -- even though they're high risk pipes, they have no known degradation.

So, they leverage the amount of inspections based on that sensitivity. So, the ASME process for doing these inspections allows you to choose the welds that are going to be in your sample. You do an inspection, and then every interval you inspect the same sample of welds. In the rule, which you'll see Joey will talk about later, we are allowing the licensees to leverage their existing ISI programs however they can to try to meet this ten percent sample.

MEMBER PETTI: So, how many welds is ten percent, rough number?

MR. RUDLAND: Well, if we talk about all of the welds, for instance in a PWR you're talking about the hot leg, cold leg, crossover leg, and the welds in between, so it can't be more than I would say 30 welds, 40 welds probably. We're talking about a similar amount of welds in there. There's also the -- and the dissimilar metal welds also.

I would assume it's about the same in a BWR. But again, this category of weld is in their risk informed program, so they can modify their risk informed program to make sure that the ten percent requirement here is reflected in their ISI programs. Okay, let's go to the next slide. So, there was a minority review on this decision.

And it was very important that we talk about this during the development. We had some discussions about this, this was not an issue that was -- that rose to the level of a non-concurrence, or a DPO, but it was something that we thought was important to document, and let the committee know that we had these discussions, and that there were some staff that felt more was needed than what we have in the rule right now.

The staff member that had this view was really concerned about these SCC mechanisms that have long surface flaws that can grow in low toughness material, and they were worried that it impact leak before break, that you could actually get break before leak, and that we need to look at more than just ten percent basically of the welds in this category.

But it's very 0plant specific, depending on the plant itself. And so, what the staff person wanted, was wanted to be able to not only inspect ten percent, but as I mentioned, ASME allows you to inspect the same sample every interval. He would have preferred that a different sample be chosen every interval, so that by the end of a 40 year life you would have inspected all of the welds at least once in that category.

And so, that was what they recommended. The discussion of this is in the white paper, but I think the staff in general thought that might be too burdensome for the return that you got on that for that additional requirement. Because realizing that they have to have different -- that the industry has to have different structures in place, whether it's scaffolding, or whatever to inspect these things.

They'd have to have new equipment possibly for each different interval as they move these inspections around to different pipe welds.

MEMBER HALNON: Hey, John, this is Greg Halnon, can you help me with this highly unlikely but possible term up here? I mean, you can make that argument for just about everything, but could you help me with what that means from a practical perspective?

MR. RUDLAND: You're talking about the likeliness of the SCC flaw?

MEMBER HALNON: Your first bullet there, a LOCA greater than TBS highly unlikely, but possible.

MR. RUDLAND: So, again, I think that the thought was that we have some evidence that you could end up with very long surface flaws. And so, the chances that you could end up with very long surface flaws is real, but the chances of those really long surface flaws leading to a LOCA is highly unlikely.

MEMBER HALNON: So, why can't we just stop there? Why do we have to go but possible? Because that to me just looks like it's someone worried about it without any real empirical and or analytical data that says that.

MR. RUDLAND: I think that was just the

view of this particular staff member. I mean again, it's based on, I'm sure, experience that we've seen. Especially since we know we have seen kind of long cracks like I mentioned earlier, in smaller diameter pipes. That does not make it impossible that it would happen in larger diameter pipes greater than the TBS.

MEMBER HALNON: Yeah, I lived through the V.C. Summer, I know that's dissimilar, but after looking at how bad that weld was, and how much crack actually went through wall, I just don't see leak before break not being real.

MR. RUDLAND: Let's -- the thing about the V.C. Summer flaw that you're talking about is a little bit different, that's an axial flaw, it's limited in growth, length by the weld itself. There was a circumferential flaw that occurred at that location, but it was over near the cladding, and so it was actually stopped by the carbon steel material.

So, it could not have grown deep, because it grew into non-susceptible material. So, that's a little bit different to talk about. There were some cases, Wolf Creek had some very large surface cracks found on their pressurizer surge nozzle back in the early 2000s also, where we saw I think it was three relatively large surface flaws in that.

So, it is possible that they could be there, but I think it's less possible that they could cause alarm.

MEMBER HALNON: Okay, we can move on, I just wanted to make sure I understood where you were.

MEMBER PALMTAG: This is Scott Palmtag, one of the things you said is this depends on the plant, can you elaborate on that?

MR. RUDLAND: Sure. So, a lot of times the cracking is very not just even plant specific, but also weld specific. So, how these welds are made, if they're repaired, and such like that is very specific to that particular situation.

MEMBER PALMTAG: So, it's like the chemical part of the weld, or the material part of the weld.

MR. RUDLAND: It's the stresses that are imparted due to the weld. So, if you have a weld repair at a location, especially if it's on the ID, you can end up with very large tensile stresses that could drive these cracks.

MEMBER PALMTAG: Wouldn't you be more susceptible to cracking once you have more sections on that, rather than just rely on a ten percent?

MR. RUDLAND: Yes. And so, usually the

ten percent, which you'll see when we talk about the rule, is focused on those that are the highest susceptible welds. So, you have to choose a sample that is representative of the ones that you expect to crack first. So, look at the ones with the highest stresses, maybe they have the highest cumulative fatigue factors, and things like that to help them decide which welds to inspect.

CHAIR BALLINGER: Is there any possibility of a kind of a human factors issue here where a plant decides to inspect X welds based on some criteria, and one of those criteria being how much it costs to do it, and that's certainly a piece of the decision. But is there a possibility that by only inspecting the same ten percent all the time that you run into a problem where there was a weld that was chosen, for lack of a better word, for human factors issues, cost, that might have been actually more susceptible?

MR. RUDLAND: Yeah, that's always possible.

CHAIR BALLINGER: So, I'm trying to take the devil's advocate position here.

MR. RUDLAND: That's always possible, and I believe that that was one of the rationales that the staff member used in his basis, was that you don't know where those weld repairs may be, right? Because they're not necessarily well documented in the fleet. But again, I think the purpose of doing this ten percent inspection, especially in a location where we know there is no active degradation, is to protect against anything that may occur.

And statistically, ten percent is good enough for this population of flaws to be able to find something, but it has to be systemic in nature, it has to be something that is not just one weld off, it's something that has to occur maybe in a couple of different welds. But typically, like in the V.C. Summer case, when you do have a really bad weld, they typically tend to leak, because the driving force for an axial flaw is greater than the driving force for the circumferential flaw.

CHAIR BALLINGER: So, you don't think we have anymore V.C. Summers out there?

MR. RUDLAND: I think there may be V.C. Summers out there, but again, I think they're going to be axial cracks, and not circumferential cracks, which are the ones that will lead to LOCA. I don't know if anybody else has any thoughts or comments, but let's move on. All right, so the next few are going to be talking about vessels.

So, the non-piping things. The first two that are on this screen are interacts from experience that we considered also. The first one on the left is carbon macrosegregation, so this was an issue that was founded again, in a French EPR, it was actually in the head in the Flamanville 3 unit. In 2015 it was discovered that the upper head had high levels of carbon that was due to the way that they manufactured that head, through the forging process.

It resulted in a higher strength location, very specific to the location in which the segregation occurred, lower toughness. And the data suggested that something that is segregated like this may be more susceptible to embrittlement. So, back in that time frame, the staff had done again, I'm going to use the term risk informed analysis, but it included a lot more than what those words say.

We did some work from the probabilistic fracture mechanics standpoint, we did a lot of review on what the industry was doing, we did a lot of review on what the French were doing in terms of analyzing this away. We also looked at how the forge that was creating these components supplied the U.S. with their upper heads, and looked at those particular individual plants that may have gotten those heads from that particular forge.

At the end it was concluded that the safety significance of that was negligible. Same with the quasi-laminar indications, this happened in a plant in Belgium. Ultrasonic inspections found a whole variety of hydrogen flaking mid wall thickness in one of the ring forgings in their RPV vessel. And they were, again, they were parallel to the wall, and subsurface, and almost all were axial, or slightly off axial locations.

And how do those flaws impact integrity, do they impact embrittlement? Things like that. So, in this particular case the staff didn't really do any probabilistic fracture mechanics, but we did do an evaluation, we looked at what Electrabel, which was the Belgian owner, what their analysis said. And we did some finite element analyses, and some risk things also included in there.

And really determined that it's not a potential safety case for any plants that may be affected by these hydrogen flakes. And again, both of those analyses are referenced in the white paper, and they go into a lot of detail in how those were dispositioned. You're more than welcome to take a read through if you have the time.

Okay, the next slide. As with the piping -- before we go into that there was two other items. The one on the left was not really an operational experience per se, it was actually something that was discovered during a set of analyses that were being conducted looking at ASME Appendix G, risk informing Appendix G, which is the rules for calculating pressure temperature limits.

Oak Ridge National Labs at the time, which was a contractor for the NRC, suggested that a small surface breaking flaw can produce a greater through wall crack frequency than the assumed quarter T flaw that was part of the Appendix G analysis. And this particular small surface breaking flaw was just large enough to penetrate the cladding of the vessel, or the tip, which is an inch of the carbon steel.

And in those particular cases the through wall crack frequencies were several order of magnitudes larger than assuming a quarter T flaw. So, that kicked off a very large effort across industry and at the agency to really take a look at what was going on here. It also involved folks at the ASME code, and everybody kind of got involved in trying to analyze this.

We did publish the fracture mechanics

studies, the industry did publish their fracture mechanics studies to really understand what was going on here. And what we found out was that even though there was an increase in the conditional probability of failure, the impact on the through wall crack frequency really was minimal due to the fact that the cool downs that we assumed were not very realistic compared to actual plant cool downs.

And so when you considered the actual plant cool down failure frequencies, and or the idealized cool down frequencies, through wall crack frequency was again shown to be low.

CHAIR BALLINGER: Now, would the P-T limit curves have influence on the rate of cool downs allowed?

MR. RUDLAND: The ASME code talks about what the allowables are for the cool downs.

CHAIR BALLINGER: So it is built in?

MR. RUDLAND: Yeah, it is built in. And the second one on this page on the right is the embrittlement for the vessel. I talked to the committee about this in the past, that the Reg Guide 199 embrittlement trend curve in some cases may under predict embrittlement for high fluence. That coupled with right now the way that the current embrittlement rules are written, licenses are allowed to defer surveillance capsule testing.

That's intended to confirm that embrittlement prediction from the ETC model. Those combined, the staff again did a risk informed analysis where we did probabilistic fracture mechanics, and other types of analyses that suggested that there may be a risk issue with the staff's confidence in the integrity of the RPV for a certain subset of plants for long term operation.

And so, the staff proposed a change to those rules in SECY 22-0019, and we're still waiting on commission decision on that.

CHAIR BALLINGER: I have it marked in my docket, four to five years.

MR. RUDLAND: '22, now we're going on '25.

CHAIR BALLINGER: Okay.

MR. RUDLAND: Yeah, the original plan that we had to the commission had us doing the reg basis, and developing the rule, and publishing the final rule in '27.

CHAIR BALLINGER: So, what is T zero, when did the four to five years start?

MR. RUDLAND: Yeah, and again, the study that we did kind of demonstrate that, the first plant really wouldn't be impacted, and it was basically a PTS issue for about ten years from the time that we did the analysis in '22. So, it really won't reach the fluence limit where there's an issue until maybe '32 time frame. But we still need to write the rule.

CHAIR BALLINGER: We're getting a couple of plants, one of them I'm sure is very old, that they're thinking about restarting.

MR. RUDLAND: I understand.

CHAIR BALLINGER: That doesn't have any influence on this fortified --

MR. RUDLAND: Not at this point, again, we looked at all the plants --

CHAIR BALLINGER: They did a 61A.

MR. RUDLAND: Yeah, they did a 61A, that's right. And we thought about that plant as we were writing this -- doing this analysis, so it wasn't -- it's not going to be a concern until about, like I said, about ten years. But the staff needs five years to write a rule, and get a final rule published.

CHAIR BALLINGER: Well, we know it only takes one year. We can only be heroic once.

MR. RUDLAND: To the manager sitting there. So again, we're waiting on the commission decision, but the staff is still concerned about this kind of thing. Let's go to the next slide. And as with the piping, we also have inspection frequency changes. So, along with the piping work, the industry is moving towards optimizing the inspections for a lot of the large vessels.

Many licensees have come in through our 50.55a(z), which is our alternative process for code requirements, and they've been granted approval to modify their inspection programs for the reactor pressure vessels, steam generator, and pressurizer, shell to shell welds, nozzle to shell welds, and inner radius inspections.

And the cumulative effects of those relaxations, in addition to the relaxation that we're talking about as part of this rule may have some impact on the TBS, and so we needed to make sure that we're okay with that. And the staff thought about that a lot, and qualitatively yeah, I think the staff is okay right now, that the appropriate amount of performance monitoring is occurring in these components, so there's not really a safety issue.

In each of the cases we've always -- each of the cases where we've approved these relaxations we've always tried to make sure that there was sufficient performance monitoring to be able to verify that the analyses that were done remains valid, and that if any novel degradation is found, it can still be found through those performance monitoring inspections.

So, what's the impact on the TBS? Again, just like with the piping, some of the issues have been analyzed through our 504 process, and are well documented, and we did consider these cumulative effects. The only thing that seems to be open at this point is this embrittlement issue, and the staff will take a look at that once the commission has taken action on the rule making final.

CHAIR BALLINGER: But it's pretty clear, the outcome on that rule is pretty straight forward.

MR. RUDLAND: I don't know what you mean by straight forward, the commission has not acted, so I don't know what --

CHAIR BALLINGER: The new ETS, the new correlation, and the --

MR. RUDLAND: Yes, we know what we proposed, our proposed action is relatively straight forward. Okay, let's move on to the next slide. And the last thing the staff looked at was secondary side piping failures, and again as Rob pointed out earlier, this is failures that may happen at piping that is smaller than the TBS impacting and causing LOCAs in piping greater than the TBS.

And it's definitely felt that current regulations did a pretty good job of making sure that that doesn't happen. GDC-4, and SRP 3.6.2 really provide assurance that that's maintained. They have to do certain things to make sure that those pipes don't pipe whip, cause any issues in that case. The piping is also very flaw tolerant, a lot of it is stainless steel, and the stainless steel is extremely flaw tolerant.

So, the probability of enough damage from the one pipe causing damage on the piping greater than the TBS is small. But guidance is still going to be needed to cover any particular plant specific issues that may impact secondary site failures, and so we'll be generating that guidance as part of the DG-1428, which we'll talk about in January.

Go to the next slide. So, that's the end of the qualitative work that the group did to look at the impacts of the Tbs. I'm going to move to the quantitative stuff. So, in thinking about how we were going to try to handle this in a relatively short time frame, one of the staff initially recognized that a lot of work had been done, at least for the PWR pipings to understand the LOCA frequencies through our leak before break efforts, and the PWSCC efforts.

Both the agency as well as EPRI have developed independent studies where we've done analyses to try to look at what the impacts of PWSCC are on those particular pipes that have been approved for LBB. There has been reports published on that, but the bottom line is that the probabilities of these large breaks, the probabilities of these large pipes breaking is extremely low.

And even especially, since like Ron pointed out, most of them have been mitigated. The concern about these large break LOCAs being a high probability event is gone. But we wanted to try to confirm, so we wanted to leverage that knowledge that we got, that was developed over the last ten years, with making sure that we could confirm the LOCA frequencies that Rob showed earlier for the individual base cases.

Again, our idea was not to recalculate LOCA frequencies, it was just to see, to run some analyses to see if we thought that the LOCA frequencies that were in 1928 were representative, or conservative, or how were they related to the analyses that we did. So, we looked at those four base cases that are shown here, we had a 30 inch RPV nozzle, a 10 inch surge nozzle, 28 inch recirc line, and a 12 inch recirc line.

And we recalculated the LOCA frequencies using the xLPR probabilistic fracture mechanics. Let's go to the next slide. So, the approach shown here, we ran all these cases out to 80 years of operation time, realizing that the 1829 work was only for 25, 40, and extrapolated to 60, we wanted to run them out to 80 years.

We chose sample sizes, and sampling sufficient to give us confidence in a ten to the minus six probability event. Which again, didn't have an extraordinary amount of time to put on a whole bunch of cases, so we had to use some optimization to run these in a reasonable amount of time. We also enabled leak detection, took credit for that as part of these analyses.

We looked at probability of leakage, probability of rupture, and leak rate, and then we took that and post processed it to calculate component level LOCA frequencies, and made those into annual frequencies by taking into account the years in which they were calculated. For PWRs, we assumed that the most relevant mechanism was PWSCC.

We looked at a couple of different residual stresses, the inputs that we used for the PWR were very similar to those cases that we used in the effort that I talked about earlier, the effort that was done to look at PWSCC right before break. We had two residual stresses, one was the residual stress to mimic the profile that was found in the V.C. Summer case that we talked about earlier, we assumed that was very conservative.

And then we used a generic representative weld residual stress profile that was developed through the xLPR program for these pipes, for both the reactor coolant loop piping, and the pressurized surge piping. For the BWR we had to use an IGSCC crack growth rate, but xLPR currently doesn't have a built in IGSCC crack growth rate.

So we had to use the generic crack growth rate model, and put in the parameters that matched the most recent ASME Section 11 IGSCC crack growth rate. And for the residual stress, we didn't really have residual stresses for the BWR welds, and so we generated them from this program to be able to run these analyses, we generated them as part of this effort.

Let's go to the next slide. All right, so the next couple slides I'm going to show some of the results. These are mainly just for comparison purposes for you. There's a lot more detail again in the reports I talked about earlier. But in each one of these plots, the Y axis, vertical axis is the annual LOCA frequency.

In this particular case it's a small break LOCA frequency, and on the horizontal access is the time in years. For most -- for all of the cases that we ran for this effort, we got no large break LOCA frequencies, which means that it's well below the ten to the minus six value. So, for the number of realizations that we run, we got no results.

In some cases we only got small break LOCA results, in some places we got small break and medium break LOCA results. But for most cases we could not get any results for the very large break LOCA. In each one of these plots the green lines, the vertical lines are the NUREG 1829 results, and the red or orange lines are the PFM results from this effort.

And for both the PWR-1 and PWR-2, you can see that the results are either representative, and kind of go through the middle, or conservative compared to the 1829 results in the LBB readings.

DR. BLEY: This is Dennis Bley.

MR. RUDLAND: Yes.

DR. BLEY: Just looking at your red curve from xLPR, it kind of flattens out, does that imply some kind of repair mechanism, or arrest mechanism going on?

MR. RUDLAND: No, we didn't assume any repair mechanisms in these particular analysis. I think the trend that's shown there is because of how the annual frequency was calculated. So, xLPR runs in a time base, and at each time step, if you divide the number by the years that you're at that particular time step, at the larger years you'll get slightly lower numbers.

So, the trend that you're seeing is due to the fact that we have normalized by year over the course of the analysis.

DR. BLEY: Okay.

MEMBER HARRINGTON: Dave, this is Craig. On the PWR-2s, are the step changes inspections, or?

MR. RUDLAND: No. So again, as I said earlier, we chose a sample size that was expected to give us a reasonable numbers at 1E to the minus six. However, you can see in those cases we're down to 1E to the minus eight, so it's due to the sample size. It's just coarse because of the sample size. It would smooth out if we were able to run a number of samples that were down to ten to the minus eight.

But again, at that particular point it didn't seem to be necessary because it shows even at that coarseness, that it was below the 1829 numbers. Okay, let's go to the next slide. Similar results for the BWRs. Everything is the same, except in this particular case there's green lines and blue lines. The green lines are the small break, the blue lines are the medium break LOCAs.

And you can see the same kind of results, you see that the lower end of the distribution for cases that Rob showed earlier, and these were indicative of the fact that the numbers that were generated from 1829 for at least the xLPR ones are demonstrated that they seem to be slightly conservative for LOCA frequencies.

Okay, let's go to the next slide. All right, so in addition to the xLPR study that we did, we also did this international operational database study, and the purpose of this was to try to do a Bayesian update on the LOCA frequencies that we had back in 2004ish time frame. The basis of those numbers that were chosen by those particular panelists that decided to use the operational database of course was only for the data up to that time of 2004.

And there's been a lot of changes in the 20 years since then, we have more operational experience, and we have new mitigation techniques, and we have better inspection techniques, and so the idea was to take the fact -- of the change -- are there any changes in the operational database that would cause us to want to change the LOCA frequencies.

The plot on the right just kind of demonstrates how much time in EFPY has changed between 2004 and 2024, so there is 952 years difference in there in terms of the amount of time that was in the operational database. So, again, the objective of this was just to reevaluate the LOCA frequencies, let's go to the next slide. The procedure was like this, where originally they calculated the piping failure precursor frequencies from an operational database for up to 2004, and then up to 2024.

The database that was used, Rob mentioned it a little bit earlier, it is the CODAP database. That is a database that is managed by NEA, and it contains all of the events, and experience worldwide on operation data, whether it's inspection defects -- leakage, defects, or failures, it's all in that particular database.

And for your information, Ron, the information that we used at the last briefing where we showed the data, it came from this database. So, from that precursor data, we were able to calculate the conditional probability of failure from the NUREG 1829 LOCA frequency estimates using the 2004 precursors. And then from that we updated the LOCA frequencies using -- make an estimate of the CFP and updated LOCA frequencies from the 2024 failure frequencies.

Okay, let's go to the next slide. This shows those estimates. So, if you look for both the PWRs and BWRs for those LOCA categories that were considered, the LOCA frequencies, the mean values of the plant level LOCA, annual LOCA frequencies went down an order of magnitude, or two in some cases, suggesting again that the full LOCA frequencies are actually better than what was estimated in 1829.

For both BWRs and PWRs the trends are about the same. And that was attributed to again, like I mentioned, improved mitigation techniques, better inspection procedures, and honestly the proactiveness of the industry to handle operational experience as it comes along. Any questions on this?

CHAIR BALLINGER: So, I look at those numbers and, again, I don’t -- probability is not a one or a zero but it's pretty low.

MR. RUDLAND: One in a billion, four in a billion, right? Yes, they're, relatively, relatively absolutely low, yes.

But, okay, so, let's go to the next slide.

Before we move on to 1903 I just want to make a comment that the staff have made the tentative decision that if the LOCA frequencies were determined to be less than what they were calculated in 1929, that we would take no -- we'd make no path forward to change the transition break size.

We felt that this confirmation effort was just to make sure that the assumptions that were made in the 1929 were reasonable.

We have not been able to do a full, complete, comprehensive study of redeveloping the LOCA frequencies. So, we didn't want to make any changes unless they showed increase in the LOCA frequencies which we see.

And so, what you'll see in the rule for the transition break size is the same as what Rob talked about earlier.

All right. So, in addition to the 1929 work, we wanted to do a reconfirmation of, of the 1903 results. And as Rob talked about, 1903 addressed seismic effects on TBS.

We looked at, we looked at three, three cases: the unflawed piping, the flawed piping, and the indirect piping failures. It used the Lawrence Livermore National Lab Seismic Hazard Curves for that particular assessment.

And as Rob pointed out, the decision from, the conclusion from that effort basically with the direct flaw -- unflawed piping had failure probabilities extremely low. So, we really didn't look any further at the direct unflawed piping, but they were definitely less than 1E-05 per year, which was the basis that was used as a basis to establish the TBS.

All right. So, where this effort -- let's go to the next slide.

Again, for this effort we, we didn't have an exorbitant amount of time to go through it and make analyses the way that was done in 1903. So, what we decided to do was take a look at the updated hazard curves, both the, you know, the central and eastern U.S., reevaluate their seismic hazard curves after the Fukushima accident. And that NUREG contained information on what those were.

And so, we wanted to be able to take those new hazard curves and see if there was any impact if we took those hazard curves, calculated the 1E-5, 1E-6 probability to see if there was any impact on this.

So, for the direct flawed piping, the associated stresses relating to those 1E-5, 1E-6 probability of exceedance events for seismics were very large. And as Rob showed, the probability for pipe breaks larger than that were shown to be lower than the TBS frequency criterion.

For the indirect piping failure, the mean probability of failure at the lowest capacity component was less than 1E-05.

I'll show some results of that in a few, in a few minutes.

And so, for both direct and -- direct flawed piping and indirect, the numbers were much less than 1E-05. And all of those results are detailed in the White Paper for Continued Applicability of 1903, which is at the ML number as shown.

I'll go into a little bit of detail next.

The next slide.

For the direct unflawed piping failure we used the same steam that was used in 1903. Basically you take the design stresses, which is the normal stresses and the SSE stresses.

You take the -- you determine the PGA for both 10 to the minus 5 and 10 to the minus 6 earthquakes. In this case we didn't take them from the LANL -- from the Lawrence Livermore results, we took them from the updated seismic hazard curves.

Basically, calculate and determine the scale factor that Rob talked about earlier which allows you to go from design stresses to realistic stresses.

And then we scaled those stresses to the higher earthquake levels.

We then normalize those results by the design stress intensity and obtain the probability of exceedance from the seismic hazards.

Go to the next slide.

This shows an example of the direct unflawed piping failure results as a function of stress at the bottom. Basically the 1, 2, 3, 4 are one times the design stress intensity, two times, three times the design stress intensity. And last is the probability of exceedance.

And the blue line is the 1 percent probability of failure numbers. And you can see at that page they're all below the 1E-5.

Taking that, taking those values, we go to the next slide.

Here is the scatter plot that shows all of the cases analyzed in this effort. And that threat probabil -- threat piping failure probability per year as a function of the number of plants, basically.

And you can see they're all below the 1E-5 per year.

They also did -- if you go to the next slide -- indirect. We wanted to recreate the indirect also. And these, again, are pipe ruptures caused by failures of major components, or component supports as a result of an earthquake.

We used the fragility curves that were in 1903 and did the same kind of analyses. And the results show that the main probabilities are less than 1E-5.

You can see the actual results on the next, the next plot.

Let's go to the next, the next slide, please.

And, again, this is the indi --

Is there a question?

CHAIR BALLINGER: I don't know. That's just some somebody that hasn't muted themselves.

MR. RUDLAND: So, the vertical access again is indirect piping failure probability as a function of site. A number -- and you can see all the numbers, again, are below 1E-05.

So, the staff felt pretty confident that the results in 1903, which were basically that the seismic effect wouldn't impact TBS, was still, was still valid now.

All right. So, one more slide.

So, we did all this work, again, within, within since about March. And staff felt pretty comfortable after going through all this that the LOCA frequencies are applicable only if, again, plant specific applicability is demonstrated. In both the 1903 results, as well as in our results, our finding elements, our PFM analyses. Again, affirmed generic loads. We didn't do any class specific loads.

On the seismic efforts are now bounding. We do believe that plant specific applicability needs to be done.

Okay, like I was saying, the LOCA frequencies and the TBS are applicable only if plant specific applicability is demonstrated. We felt that both the results from the original efforts as well as now were not really that bounding, they were more representative. And so, plant-specific applicability is key to do the operation plan in 1903.

More importantly, we felt that if they were not able to demonstrate that they were applicable, that, for instance, if it was a redesign, that they could develop their own transition break size using the methodologies that we did, that we've done in this set.

The change in requirements was that the staff really felt that the inspection of these piping welds with diameters greater than the TBS was needed to ensure the LOCA frequencies remain applicable. We wanted to be as flexible as possible on that, while still retaining those inspections. So, allowed the individual licensees to leverage their ISI programs to account for those inspections as, as they need to.

CHAIR BALLINGER: Can we go back to 67?

MR. RUDLAND: I think so.

CHAIR BALLINGER: I was wondering why that scale factor was pretty large, 1.4, 1.5 and thereabouts?

MR. RUDLAND: Well, we're scaling from, we're scaling from design conditions to more accurate conditions. That's not unreasonable. I'd expect --

CHAIR BALLINGER: It's not unreasonable. It just seemed large.

MR. RUDLAND: And the design loads aren't, the design stresses aren't large compared to --

CHAIR BALLINGER: So, do those probabilities that you show scale linearly with that scale factor?

MR. RUDLAND: That's a good question. Let me, let me call a friend on that.

Se-Kwon, come up to the -- see where the green light is there.

MR. JUNG: Yeah. I do the analysis that, as Dave mentioned, that I take the stresses more actually linearly scaled up.

CHAIR BALLINGER: Okay.

MR. JUNG: Yes.

CHAIR BALLINGER: All right. Thanks.

MR. RUDLAND: And that was Se-Kwon Jung from DEX, engineering conditions. He helped with all the seismic analysis.

CHAIR BALLINGER: Thank you.

MR. RUDLAND: Okay. If we can go back to the last slide, last slide?

Yes. I think that was it for me.

Again, if there's any questions, happy to answer them. If not, I think I turn it over to Joe.

So, we would request a 5-minute break, bio break.

MEMBER KIRCHNER: Ron, would you entertain another question?

CHAIR BALLINGER: How imminent is this break required?

MR. RUDLAND: We can handle the question until we do what he needs to do.

MEMBER KIRCHNER: Okay. This is Walt Kirchner again.

So, just to reflecting upon the one opinion of your staff about inspections, do you feel that industry-wide, Ron had a line of questioning earlier, you know, cost considerations come into play in inspection choices? Also, dose and exposure considerations play. And then degree of difficulty in terms of access and such.

So, based on your experience looking at the plant, do you, do you see any areas -- and I'm thinking mainly PWRs now -- where there are welds that are not being inspected that should be?

You mentioned there's on the order of 30 major welds --

MR. RUDLAND: Yes, yes.

MEMBER KIRCHNER: -- in a PWR plant. Are all 30 of those being covered either --

MR. RUDLAND: Yeah. Now, some of them have never, some of them have never been inspected. You know, they were inspected at -- during construction and during the pre-service inspection. But they're probably not part of sampling.

All of them cannot be part of the sampling program because it is a sampling program. So, you know, the typical, I mean sampling program for piping is, for Class 1 piping is 25 percent, just across the board 25 percent.

But these risk informed programs which allow the plants to inspect only those welds that have high risk is really only a 10 percent sample.

So, there are going to be welds out there that have probably never been inspected. It's going to be very plant specific how many welds. These welds that we're talking about, it's going to be very plant specific on whether or not those welds are currently their risk informed program or are being sampled from -- they're being inspected within the risk informed program or not.

MEMBER KIRCHNER: Okay.

MR. RUDLAND: They're all part of the program, but whether or not they're sampled or not is, is very plant specific.

MEMBER KIRCHNER: Yeah. I understand all that. But what I'm asking, I guess, more specifically is there, in the current fleet is there a weld location that isn't being sampled in any of the PWRs?

MR. RUDLAND: Not in, not in the reactor part of the boundary, no, the reactor coolant building, no. They're all --

CHAIR BALLINGER: Okay. That --

MR. RUDLAND: They're all part of some inspection program, whether it's the new inspection program, or it's the former inspection program, or an augmented inspection program.

MEMBER KIRCHNER: Okay. No, you gave me the answer I was looking for.

All right, thank you.

MR. RUDLAND: But that doesn't mean every single weld is inspected.

MEMBER KIRCHNER: No, no. No, I didn't mean to imply that. I just wanted to hear you say that all welds -- no, let me try to pick my words very carefully here -- that if there are on the order of 30 or 50 major different weld locations, that these have all been surveyed and inspected at some point?

MR. RUDLAND: They're all part of the sampling program. Whether any one particular weld has been inspected over the 40 years is unknown.

So, there is a possibility that there are welds out there that have never been inspected.

MEMBER KIRCHNER: No, I understand that part.

MR. RUDLAND: Okay.

MEMBER KIRCHNER: I'm just thinking generically for a PWR there aren't -- that they've covered all the weld locations in the generic sense.

MR. RUDLAND: All the welds are basically part of their -- in their program.

MEMBER KIRCHNER: Yeah.

MR. RUDLAND: We don't know. I think Walt's getting at generally, you know, do we think if there is a weld in, for instance, in a hot leg that's a common location --

MEMBER KIRCHNER: Yeah. That's inaccessible or something.

MR. TREGONING: Let me use Westinghouse plants. If, if some plant, some Westinghouse 3-loop plant is inspecting that weld, we don't know that because, --

MEMBER KIRCHNER: We don't know that.

MR. TREGONING: -- you know, the ISI programs were developed on a plant-specific basis.

Right? And, you know, based on their, again, there's generic categories of welds that break down that become part of their inspection program.

But then they're supposed to tailor what loads they look at based on unique operating experience and conditions within those, within the particular plant. So, they don't coordinate in some way to make sure that, okay, somebody's inspecting this weld in this plant.

That level of coordination is not only not required, it's, it's not done to my, you know, to my knowledge.

But realizing you've got, like you said, Walt, there are a variety of --

MEMBER KIRCHNER: Yeah.

MR. TREGONING: -- incidences that cause them to choose which welds they're going to inspect. Every licensee has their own set of rules to inspect.

So, it's a good possibility that a good number of them are being inspected, but I can't guarantee that all of them have been inspected.

CHAIR BALLINGER: But the point is, is that you may not know, but the data's there.

MR. RUDLAND: The data's there.

CHAIR BALLINGER: You think the data's

there.

MR. RUDLAND: There probably is an old employer probably has a lot of the data.

CHAIR BALLINGER: But the critical question might be these welds are all ranked in some way to susceptibility.

MR. RUDLAND: Yes.

CHAIR BALLINGER: Is there the possibility that a highly-ranked, that is to say a bad one, has not been inspected, is one of that population that has never been inspected?

MR. RUDLAND: So, again, how they're ranked, at least in a risk informed program, is once the, once the risk significant, if that thing fails, that's the first ranking.

The second is what's the degradation on that?

So, if you have a high risk, low/no degradation, it's a low category weld, right, even though it maybe has a high risk of failure.

Which is why we are concerned about those ones that are greater than the TBS. If those are in that situation they are high risk welds, low because of the no degradation. And so, they get less inspections.

MR. TREGONING: But just to be clear, you've got risk informed program, it tends, if we're talking about Class 1, from a consequence perspective they all have basically similar consequences in that run of pipe.

MR. RUDLAND: In that run of pipe for Class 1. So, they're really based on material pipe susceptibility.

So, if you're talking about a stainless steel weld, it doesn't matter what weld that, you know, if it's in a heat remover, safety injection, or the main coolant loop piping, if it's a stainless steel weld it's Class 1. It's likely all part of that same inspection population.

And then they will choose their 10 percent sample size based from that population. And if they're all equally susceptible then, yes, cost, acceptability, ALARA, all of those things factor in because, obviously, if you make the presumption that they're all equally susceptible, you're going to want to try to minimize those other costs and risk factors associated with the inspection.

CHAIR BALLINGER: So, but somebody that would want to use this option, at that point the data would -- you would find out.

MR. RUDLAND: Yeah.

MR. TREGONING: Yes.

And to Walt, I want to comment to Walt.

So, for instance, in some of these augmented programs like the certain amount of weld programs, so there's a very comprehensive program for inspection and mitigation of a certain amount of welds. Because of that operational experience and the fleet-wide behavior those welds are all inspected. Okay?

So, 100 percent of those welds are inspected or have been inspected. And even if they're, or even if they're mitigated they're in, within some kind of inspection program. So, because we know the susceptibility of those materials, the augmented programs are focused on those and they get, they get the inspection.

So, if we did find out some new mechanism popped up, augmented inspection programs would, would probably find those.

CHAIR BALLINGER: But in effect you have, basically, a canary in the coal mine because there are some plants that are not mitigating these.

MR. TREGONING: Yes.

CHAIR BALLINGER: But they have the

augmented inspection?

MR. TREGONING: They have. They are inspecting it more than those that aren't.

CHAIR BALLINGER: They're inspecting it more. So, they are in one sense leading.

MR. TREGONING: Yes.

MEMBER KIRCHNER: Thank you, Dave. That's enough.

I just wanted to get that on the record.

MR. RUDLAND: You asked the question, Walt.

MEMBER KIRCHNER: Thank you.

MEMBER PALMTAG: I have a very simple question.

Could you go back to the graph with the LOCA versus the time, maybe it was 65.

MR. RUDLAND: So, back in the PFM section, I think.

MEMBER PALMTAG: Yeah. So, I'm not a fractions mechanics person but.

MR. RUDLAND: No, yeah, keep going. Keep going.

MEMBER PALMTAG: That's good enough.

Can you tell me why the LOCA frequency goes down with time?

MR. RUDLAND: I talked about that a little bit, a little bit earlier.

So, again, these LOCA frequencies are annual frequencies. The way that problems with fraction mechanics runs is that it runs on a time basis; right? So, the way that we calculate the annual frequencies, we take the time step that we're at and we divide that by the year that we're at a particular point.

So, if the LOCA frequencies aren't changing much then they're going to look like they're going down because dividing by 80 is a lot, is a lot smaller than dividing by 20.

MEMBER PALMTAG: It's just more of a normalization?

MR. RUDLAND: More of a normalization. When you mate LOOP to annual basis.

MEMBER PALMTAG: Right. Okay.

MR. RUDLAND: Yeah. The problem is with fraction mechanics kind of do a cumulative fre -- cumulative probability is not what they calculate.

MEMBER PALMTAG: It would be kind of like a bathtub shape formation if you --

MR. RUDLAND: It won't go back, it won't go back up, right, because as that number gets bigger it's going to keep, keep looking like it's going down unless the absolute value of the frequency increases.

MR. TREGONING: So, just to be clear, they're going down because you're getting no new realization checks.

MR. RUDLAND: Yes. That's right.

MR. TREGONING: You're getting no ruptures in the scenario.

So, when you saw those vertical lines, that means you've got a least one realization that had a rupture. So that's why the frequency jumped up in that particular time period.

These that are showing this gradual decrease you're getting no new realizations, and then the frequency is just decreasing because you're accumulating more years without getting a rupture, if that makes sense.

MEMBER PALMTAG: Yes. Thanks.

CHAIR BALLINGER: Okay, we are, just to renormalize us, we're scheduled for a break at 2:30. It is now 2:24. In an effort to get us behind is it appropriate to just keep going or should we take the break.

MR. RUDLAND: Joey doesn't need a break.

CHAIR BALLINGER: And I don't know how much time, how much time we need for Joe's presentation.

MR. MESSINA: I think it's supposed to be scheduled an hour-and-a-half; right? I don't know if you'll take that long.

CHAIR BALLINGER: Okay. I'm fine with keeping going. But there are other members that might disagree with me on that.

Do we have a disagreement? We'll observe the 5-second rule.

Let's go.

DRAFT PROPOSED RULE LANGUAGE FOR 10 CFR 50.46A

MR. MESSINA: Okay. So, again, I'm Joe Messina, reactor systems engineer in NRR. And I'll be walking through the draft proposed rule language for 50.46a, and a reminder that this is a voluntary alternative to 50.46.

Next slide, please.

So, to provide outline of the rule, several sections I'll go through. And I'll talk about each of these as we go through. But sometimes it's nice to have this outline to more easily parse the rule.

Next slide, please.

So, in the definition section there's a bunch of definitions. I've highlighted some of them here that might be interesting or used throughout.

And so I'll start with changes enabled by this section means changes that would meet 50.46a but would not meet the current 50.46.

Entity would mean, you know, the application, holder of a construction permit, operating license, COL, SDA, manufacturing license, or applicant for a standard design cert rule.

LOCA means the coolant system LOCA here with some slight modifications from the current version in 50.46. First we, at the end we included the LOCAs at or below the transition break size or just design basis accidents. We clarify that.

And that LOCAs larger than the transition break size are beyond design basis accidents.

We also, current 50.46 definition of LOCA has, it says in the first bold, bolded section there it says breaks in pipes in the reactor coolant pressure boundary. We made it breaks in the reactor coolant pressure boundary, partly to align with the, align with the definition of LOCA in the GDCs.

The transition break size, as we talked about, is the same as Rob talked about. So, I don't believe I need to go through that.

But it's listed here. And the date that we listed here, I'll explain why that's important in the coming slides.

MEMBER PALMTAG: Sorry. Can you go back to -- Thank you.

(Pause.)

MR. MESSINA: Okay, next slide, please.

The paragraph (b) of 50.46a, is a lot of words in paragraph (b) if you look at it. But I tried to summarize it here that it's mainly operating licenses issued prior to this date, December 31st, 2015, and entities whose design is demonstrated under paragraph (c)(2) to be similar to the design reactor -- the designs of reactors licensed under part 50 before that date.

And the reason for us having this date is that the NUREG-1829 frequencies in the transition break size were based on plants licensed before that date. LWRs licensed after may have different piping materials, configurations, operational and service conditions among, among other factors that may impact the piping break frequencies and, thus, the transition break size.

I'll talk about paragraph (c)(2) here since it's mentioned. I'm talking about it here instead of later.

And paragraph (c)(2) states that for

plants after the date, an analysis should be submitted that demonstrates why the proposed design is similar to reactors licensed before that date, such that provisions of 50.46a may properly apply.

And it must include a recommendation for an appropriate transition break size and a justification that the recommended TBS is consistent with the technical basis of this section.

MEMBER PALMTAG: So, practically, the plants after would be in P1000 NUSCALE?

MR. MESSINA: Yes. And any other LWRs.

So only, yeah, only those plants in 1020 NUSCALE.

MEMBER PALMTAG: Okay.

MR. MESSINA: Next slide, please.

So, a lot of discussion was had previously on inspections. On inspections. And this is the wording in the rule.

So, a licensee must inspect 10 percent of the welds, circumferential welds in a PWR that have the highest failure potential.

And this requirement is new for the 2010 draft rule, added in part in response to the public comments to original 2010 rule to reduce the burden of demonstrating plant-specific applicability, as well as to verify that analyses that predict component failure remain accurate through the time the component is analyzed, so that it captures any new degradation phenomena that may potentially occur and have an effect on LOCA break frequencies and the transition break size.

MEMBER HARRINGTON: This is Craig.

So, the focus concern is more on the larger side. Are you assuming that other mechanisms will manage to keep inspections in place at or below TBS?

MR. MESSINA: Since we're not, since we're not redesignating the -- we're not touching the LOCAs below the transition break size, we're not really --

MEMBER HARRINGTON: Just letting it be?

MR. MESSINA: Yeah, we're letting, leaving them alone.

MR. RUDLAND: These inspections that he's talking about here are really just focused and need to assure that we catch something that's not all different; right? And that the analyses that we had to say, continue to say that.

In the statement about "reduces the burden of demonstrating," so one of the comments we had -- and, I'm sorry, Joe, if I'm stepping on your toes a little bit -- one of the comments we got back originally was that the plant-specific applicability guidance that we drafted at that time was way too burdensome. And the industry was, like, that's just too burdensome.

So, what we did here now is allowed them to program inspections and then reduce the burden in the plant-specific applicability, which we'll hear about in January.

MR. MESSINA: Okay. Next slide.

50.46a is on application, so information that the licensee or entity that is wanting to adopt 50.46a, we need to provide:

An evaluation of applicability of the transition break size;

Weld inspection report;

Description of the risk informed evaluation process to analyzes -- analyze change made, changes made under this rule;

Description of how they are evaluating the continued applicability of the transition break size;

Description of any non-safety systems that are credited for LOCAs greater than the transition break size. And they would need to be placed in the Tech Specs.

And now this, I'll talk about this a little later, but above the transition break size we would allow crediting of non-safety systems just as long as they're placed in the Tech Specs.

You'd do an evaluation of the leak detection program.

And same as I said before, for reactors licensed after the 2015 date, an analysis demonstrating that the reactor design is similar to those before the 2015 date and the appropriate transition break size.

I appear to have also, I forgot one bullet. Paragraph (c)(3) is a statement that says that the ECCS performance must meet the criteria in paragraph (e). So, that should be obvious, but.

Next slide.

Paragraph (d) is the programmatic requirements, so, the requirements that licensees would need to maintain throughout use of this rule. So, similarly, this is similar just as we progress. So, that's it.

ECCS models and methods are maintained and meet the requirements in paragraph (e);

Lead detection equipment are available;

Changes are evaluated in a risk-informed evaluation;

And then this fourth item was in the rule language that the ACRS was sent a month ago, but has since been removed during concurrence. And this had previously stated that risk assessments must be maintained and upgraded at least once every 5 years.

So, that was removed during concurrence. And you'll see later on there was also a reporting requirement associated with this later in the rule that was removed as well.

The fifth item here is that it must be demonstrated that any changes made do not invalidate the transition break size.

And No. 6, during operation, licensees must perform the weld inspections every subsequent in service internal, so to identify any additional degradation, as we had been discussing. So, that would be, you know, about every 10 to 12 years.

Next slide.

Paragraph (e) is on ECCS performance.

(e)(1) establishes two principle ECCS acceptance criteria that the fuel remains in a coolable geometry and that there's -- that decay heat will be removed in the long term.

So, this maintains coolability and removal date decay heat as fundamental objectives that really have been there since the origin of 50.46.

And this, the structure of (e) and some of the other paragraphs such as (f), similar to those in 50.46c, the draft final rule. 50.46c had two principal ECCS acceptance criteria, but the first was core temperature rather than coolability. We changed it to coolability in light of how that, how coolability remains fundamental to LOCA analysis since 19 -- the 1970s.

CHAIR BALLINGER: This is Ron now.

This is an interesting change. Because if you look at the earlier ECCS, coolability was assumed to be satisfied if you maintain the 2270 percent.

MR. MESSINA: Yes.

CHAIR BALLINGER: But now it's, so, coolability was kind of an afterthought, kind of an automatic. Now it's not an automatic.

MR. MESSINA: Yes.

Next slide, please.

So, paragraph (e)(2) is ECCS performance at or below the transition break size. And most of this wording is the same, is very similar to that in the current 50.46.

We're not touching the LOCAs below the

transition break size. We still expect them to be analyzed with high probability modeling. And to date we have interpreted that to be 95-95, so.

Next slide, please.

MEMBER ROBERTS: Excuse me. Before you do that slide, what does the third bullet mean: “Changes in fuel geometry must be addressed”?

MR. MESSINA: The changes in -- So, first we have to consider, you know, the effect of ballooning. And now that -- and, you know, the fuel fragmentation and relocation into a newer region. Such as that.

MEMBER ROBERTS: So, you’re not precluding burst for the breaks below the TBS? So, if there was burst and FFRD and they’re dispersing the fuel into various cases, that’s intended to be covered with the high probability, if you hear?

MR. MESSINA: To follow the interruption, for breaks below the transition break size we would expect that licensees show that fuel did not disperse.

We discuss that a little bit in the statements of consideration. But above the transition break size we are -- we relaxed that.

MEMBER ROBERTS: Right. So, your expectation is that there wouldn’t be burst. But the rule doesn't preclude burst?

So, if you had burst in a break below the TBS, is that what that bullet point's intended to cover, essentially including burst by the way this is written?

MR. MESSINA: So, for like low burn-up fuel you can get burst and not have fuel dispersement. So that's, that's always been, we've always understood that burst would occur during a LOCA. Now we know that fuel will disperse potentially for those high burn-up runs, but not for all the runs necessarily.

MEMBER ROBERTS: All right. But the other document, at least for ALS, was that for all breaks below the transition break size there will not be burst. And that would be preventing FFRD, that would be the end of it.

I'm still wondering the way this is written, is that the intent, or would there be a pathway to actually have dispersal in a break below the TBS and use design basis methods to evaluate that potential? That was from option 3, I remember, from your initial list where you said nobody was in favor of. But that would seem to be where you would end up if you had burst.

Just an observation, if the expectation is how this is written, the implementation would be no burst for breaks below TBS. I guess it doesn't matter; right?

It seems kind of odd to leave that gap. It looks like the way, at least the way the bullet point's written there that there would a pathway to have burst and still come through the analysis.

I guess I'll leave it at that.

MR. MESSINA: Next slide, please.

So, paragraph (e)(3) starts out very similar to the paragraph (e)(2), but this is for LOCAs above the transition break size.

And like the current regulation, it says that must be analyzed up to the most severe -- most severe LOCAs must be analyzed up to the double-ended guillotine rupture of the large pipe in the RCS.

And in this is says there must be assurance to at least a best estimate that the ECCS and fuel system acceptance criteria are met.

I'll go into a little bit more about best estimate in the slides.

Changes in fuel geometry must be addressed.

Calculations may take credit for availability of offsite power.

You do not require assumption of a single failure.

And non-safety-related equipment may be credited.

So, these LOCAs above the transition break size would be beyond-design-basis accidents and they would be able to be analyzed with nominal and unbiased analyses. You know, that means no intentional conservatisms, realistic representation of plant configuration, so no non-physical operating conditions with inputs as may be done today.

And this was done in other beyond-design-basis accidents such as ATWS and station blackout.

Questions, Walt?

MEMBER KIRCHNER: Yes, thank you.

Well, I'm looking at the overall implications of this. And the first thing that occurs to me is that the double-ended guillotine break is actually the design basis for the containment.

So, are you going to talk about -- see, when you, when you make a break like this, I'm concerned about the implications elsewhere. Containment design basis usually is driven by this double-ended guillotine break. So, you're saying that up to TBS is a design basis accident, beyond TBS is a

-- I misspoke.

Up to TBS, less than TBS is a DBA. Beyond TBS is a beyond-design-basis accident. How does that impact things like containment assessment and analysis?

Because containment's usually designed for the design basis accident. I mean, that's the basis for the siting.

MEMBER PETTI: But, Walt, let me make sure I understand.

The containment response is a function of the internal energy in the primary system.

MEMBER KIRCHNER: Right.

MEMBER PETTI: The best estimate nature there is more than response of the core. You're still going to blow down all that steam, --

MEMBER KIRCHNER: Exactly.

MEMBER PETTI: -- and the temperature. And so, the containment should see this at about the same pressure; right?

MEMBER KIRCHNER: Yeah. It may change the time it sees the peak pressure and temperature, but --

MEMBER PETTI: Might change the time however?

MEMBER KIRCHNER: But I'm just concerned

that if you make a -- if you say TBS is the design basis accident threshold, what are the implications?

So, you're saying -- I'm thinking of Chapter 15, which is usually comprehensive design basis accident analyses. So, --

MEMBER PETTI: Yeah. So, --

MEMBER KIRCHNER: So, you would analyze in Chapter 15 up to the double-ended guillotine break no matter where the TBS is?

MR. MESSINA: For containment sometimes is used, I believe, a little different assumption. So, for LOCA it's most conservative that, you know, the energy of the core stays within --

MEMBER KIRCHNER: Exactly, yeah.

MR. MESSINA: And while for containment you want the most conservative would be that the energy is released as quick as possible.

So, there are some differing, differing assumptions in the methods.

And, John, did you want to say anything more on that?

MR. LEHNING: Sure. This is John Lehning from NRC staff.

And so, the changes that we're making, part of what Joe is talking about, they are for 50.46.

And those are for the core coolability requirements. And, so, there aren't any changes being proposed here relative to other requirements for containments.

And I think Member Petti's comments were quite correct that we wouldn't expect there to be a whole lot of differences relative to doing a very precise best estimate or, you know, intentionally adding conservatisms or uncertainties, because I think that really isn't a part of what exactly is being done there with those methods.

And, so, hopefully that clarifies.

Historically, payment analysis is deterministic because it's supposed to be the final barrier, final physical barrier. And if we had this discussion when I was doing this for money, you know, 20, 30 years ago and, you know, I had licensee people explain to me you'll never see that loot. You know, best estimate containment is sacred. So, we keep the methods.

MEMBER PETTI: I think I had a question on the fuel geometry for breaks less than TBS.

You still proceed if, you know, you could have analyses that push through ballooning without rupture; right? Because you could say ballooning, of course, you could very well get relocation. And, of course, that exacerbates the situation.

It's some kind of cliff edge even if it doesn't result in ballooning.

The models that have been, NUREG-630 has been the model that one's used for years, I was wondering we ought to revisit this in light of, you know, this change.

You know, I've always kind of felt like these two things kind of went together. And that the uncertainties associated with rupture and ballooning together I just really, I had never ever really thought of them separate in this way.

I don't know, I'm going to have to study this, though.

MEMBER PETTI: I think this a generic issue about, you know, we're here in 50.46 space, but this could have tentacles other places. And I know you guys have tried to think through all of those as best you can.

But there's a lot of regulation here. And operators, licensees and the like, how do you know that you haven't tumbled into something, you know, unexpected?

MR. LEHNING: And those are good comments.

And I just wanted to correct one comment that I made.

So, there is, there was a change to some of the wording on GDC 50 and which was pointed out to me. The change was to look at the containment analysis realistically rather than in some conservative way.

And I think just over time the containment analysis methods have been becoming increasingly more and more realistic. And there aren't necessarily a lot of regulatory requirements built in for conservatism. But I think was, as part of the idea that behind beyond design basis, this was made explicit in that GDC there's some language there in the Federal Register notice to that effect.

And, again, this is John Lehning from the staff speaking.

MEMBER PETTI: So, just to be clear, what analysis is looked at for the containment? Will it be the break at the TBS, or will it be the beyond design basis old double-ended guillotine break?

Which one gets used in the evaluation?

MR. LEHNING: This is John Lehning again.

I think that that could be either one. I think the most limiting one, and my expectation, is that that would be the large break in a realistic way.

Because as we were discussing before, there are not some requirements to handle. And we know pretty much this internal energy of the system pretty well. So, I would highly doubt that the smaller break, whatever conservatisms, may be the limiting one.

MEMBER PETTI: Okay now, thanks. That clarifies it.

MEMBER KIRCHNER: I'm not so certain. Yeah, for a large dry containment, probably the time it takes to blow down the entire primary system and start refilling it probably doesn't have a large impact.

But for an advanced design with a smaller containment, this is a big difference.

So, if you say -- just I'm just trying this out. Unfortunately, I'm doing it in realtime, so bear with me.

If you say, if you define design basis accidents, which is usually where we, we make the normal assumptions, single failure, et cetera, et cetera, we go through, say, fibrillated systems and so on, if you limit it to TBS, then would it be possible to design the containment for a lower pressure temperature rating than if you assumed the largest break discharging instantly into the containment?

MEMBER PETTI: Yeah, Walt, you're talking about designing a legacy plant today from scratch. The rule only applies to the reactors that have already been licensed by December something, 2015.

MEMBER KIRCHNER: Right.

MEMBER PETTI: So, I mean, unless you, are you thinking the ice condenser containments?

MEMBER KIRCHNER: No, no, no. I'm just thinking of the implication of saying up to TBS is design basis, beyond TBS is -- sorry, I misspoke again.

Up to TBS is the design basis accident, and beyond is, greater than TBS is beyond design basis accident.

And I'm just going through the, I'm going through 10 CFR 50 very quickly in my head and trying to think through all the implications of that.

MR. LEHNING: This is John Lehning again.

And I think maybe there's just some terminology issues that I wanted to point out that beyond design basis in this instance doesn't mean that it wouldn't need to be considered in the plant design. It just means that it would be consid -- it could be considered, potentially, according to less restrictive conservatisms and so forth.

But there's really three categories of things. And some other countries may do it different with design Category 1, 2, and 3 or what have you. But, in essence, this beyond transition break size, per the requirements here it would need to be considered in the design but just in a more realistic way potentially than that transition and below those ranks, if that's clear.

So, I would agree, I think, with the premise that you made that you might expect that that beyond transition break would be more, more limiting.

MEMBER KIRCHNER: Okay, thank you.

MR. MESSINA: All right. Next slide.

So --

MEMBER ROBERTS: Sorry, John. The question is kind of plant lower their design containment pressure based on this rule change?

Sorry, you said that way in the back.

MR. LEHNING: And, again, this is John Lehning.

And I think that's a possibility. But there are other breaks that -- a steam line break in containment that may limit some plants, depending on their design. And there may already be somewhat realistic methods being used in their analysis.

But I would say, in theory, it is a possibility that this could be, if that plant has a limiting break for the double-end guillotine and they have some conservatisms built in, it's possible that they could look at this change and then potentially go to a different method that is more realistic and maybe lower, at least lower the -- gain more margin that way, I guess I would say.

MEMBER ROBERTS: Okay, thanks.

MR. MESSINA: Paragraph (f) of 50.46c has fuel performance criteria. And there are four criteria here:

Address cladding degradation phenomena;

Maintain fuel coolability;

Avoid explosive concentration of combustible gas;

And demonstrate long-term cooling.

Now, these criteria, one is based from the previous 40.56c rule. So, under Item 1 we would expect, you know, a PCT limit, a local oxidation limit that are, you know, for example, currently in the rule and may have been in 50.46c final rule language.

And those guidance for addressing these criteria are in DG-1263 for Zr-U02 fuel. And, you know, licensees would address those 50.46c embrittlement research findings.

And this, these, this paragraph is fuel-technology neutral, so there wouldn’t be, need to be an exemption if, for example, more advanced fuels are used such as maybe, maybe non-U02 or silicon carbide fuel if they need an exemption to this paragraph.

MEMBER HARRINGTON: Joe, this is Craig Harrington. This is not a change that you made, but can somebody comment on the sort of convoluted wording after any calculated successful initial operation of the ECCS? What's hiding behind that statement?

MR. MESSINA: We were intending for it to mean long-term cooling. Decay heat's removed for the core.

MEMBER HARRINGTON: Just, I don't know, the -- it says calculated successful initial operation. It just -- and I mean, I know you haven't changed anything there. It's a legacy set of words, but just seems odd.

CHAIR BALLINGER: Yes, is there something that's not successful?

MR. MESSINA: If it's not successful you don't get to the next phase of the analysis, right, which is removing --

(Simultaneous speaking.)

CHAIR BALLINGER: Yes, you really don't pass go.

MR. MESSINA: Right.

CHAIR BALLINGER: Yes.

MR. MESSINA: So you fail the first criteria.

MEMBER HALNON: Yes, this is Greg Halnon. Isn't this -- have to do when you have an actual operation, you go back, analyze and make sure that there was enough flow going through it based on back pressure and everything else?

MEMBER HARRINGTON: So a retrospective look kind of thing, Greg?

MEMBER HALNON: Yes, I believe that's what this means is that if you do have a successful -- because you have inadvertent operations. I'm not going to say all the time, but occasionally. You can go back through the event reports and see that. And we always did an analysis of the flow rates to make sure that it was operating correctly. So I don't know if that's the same thing, but that kind of rings a bell for me.

MEMBER HARRINGTON: Okay. Maybe so. It just seemed odd wording.

MEMBER PETTI: Yes, when I read it I thought that first set off my commas is make sure you quench the core. And then the second is make sure you can still do long-term cooling, right?

PARTICIPANT: That's how I read it, yes. That's --

MEMBER PETTI: But I agree the words don't -- doesn't say that. You need an interpreter of it, yes.

MR. LEHNING: Joey, I could just add to that a little bit. I think it's just commenting on the fact -- and John Lehning again speaking from the staff -- that there was more of an understood or engineering nature to this long-term core cooling, at least initially. And so the thought was that these criteria in paragraph B, 2200 and so forth, that those were for the initial operation, but then by just demonstrating at least at that time that you continue to provide the flow, you continue to keep it covered. Then it's pretty much understood that you'll be able to continue to do those things now.

I mean, as time went on we've realized things like foreign precipitation and other things so it's not always that simple especially for some of the newer reactor designs. But I believe that that is the thought behind this wording that was again from the original rulemaking back in 1973 time frame.

DR. BLEY: Dennis Bley. Listening to this, I guess where the discussion started everything else I've been hearing says that demonstrate that after any successful initial operation of ECCS everything works right. But the calculated successful initial operation is the thing I think that makes everybody hang up here. Yes. But, no, at least I find confusing.

CHAIR BALLINGER: I'm just wondering why you need the words between the two commas.

MEMBER PETTI: We can --

(Simultaneous speaking.)

MEMBER PETTI: -- because there's two functions. You have to quench the core.

CHAIR BALLINGER: Okay. I got that.

MEMBER PETTI: And then you have to have long-term cooling.

CHAIR BALLINGER: Right. But --

(Simultaneous speaking.)

CHAIR BALLINGER: -- successful.

MEMBER PETTI: But there was all sorts of discussions about could you quench in the very early days and re-flood and all the hydrodynamics there and all of that contentious in the earliest days because the codes weren't anywhere near as good as they are today.

CHAIR BALLINGER: But would you calculate an unsuccessful operation?

MEMBER HARRINGTON: I think the intent is clear, but -- broadly, but that sort of suggests that if you calculated that it's not successful, then you just go on about life. That makes no sense. It's just curious.

MEMBER KIRCHNER: Could you enumerate -- could you explain further No. 3? This is a surrogate for in effect -- well, 1 percent of the zircoloid cladding inventory being oxidized.

MR. MESSINA: Correct. Yes. So the current rule has the 1 percent hydrogen generation in the rule so we're making this more performance-based and moving that to the guidance document.

MEMBER KIRCHNER: Okay.

MR. MESSINA: Because other claddings may produce possibly different combustible gases, silicon carbide, maybe it's carbon dioxide that is produced from (audio interference).

MEMBER KIRCHNER: But often assuming that you have this break in the primary system, then the implication is that your concern is either a combustible mixture in the primary system and/or the containment. I mean, will the guidance address where -- what's -- how do you determine that? It's design-specific.

MR. MESSINA: I think the guidance goes back to what's basically in the current rule. If James wants to talk more about what's --

MR. CORSON: Yes.

MR. MESSINA: James, go ahead.

MR. CORSON: Yes, this is James Corson from the staff. So I'll say a little bit about this tomorrow, but basically in the guidance for zirc-based cladding we have just the exact same limit that's in the current rule.

MEMBER KIRCHNER: Okay.

MR. CORSON: So there's no guidance for how to come up with a different criterion. If industry wanted to come up with something different, they could propose it. We could review it as we do anything else. But for now we just moved the prescriptive requirement from the rule to guidance.

MEMBER KIRCHNER: Okay. Thank you.

MEMBER PETTI: Joe, just in terms of the actual words does it say avoid an explosive concentration or does it say avoid explosive concentrations? Because there could be hydrogen gradients in the containment and there could be pockets and the like because it's lighter than -- the steam is going to condense and there's also some mixing dynamics that the codes calculate.

MR. MESSINA: I think it might be as written here.

MEMBER MARTIN: (Audio interference) on all these, the last one with -- I'm familiar with oxide from my background. When you say look at effects of crud and oxide, is kind of a deterministic sense or are you still allowing for realistic models that can treat these with uncertainty? And the crud part, I didn't think we had the science. I mean, I know we had Bearer and Castle (phonetic) and all that and I -- whether we're even there to really fully appreciate the effects of crud, maybe that's not captured in the guidance. But that's kind of new, too, isn't it? So I had kind of two questions, but --

MR. MESSINA: So this part was exactly what was basically in the 50.46c draft final rule. So generally I believe people assume a certain amount of crud is on the rod in their analyses.

CHAIR BALLINGER: Yes, I think 80 or 100

microns is usually the limit, right?

MR. MESSINA: I don't recall --

CHAIR BALLINGER: Yes, I think the heat transfer gets a little bit impacted if it's above some number, but I thought it was either 80 or 100 microns.

MEMBER MARTIN: So going back to the first question about do you treat that deterministically or is this -- it's captured as an uncertainty parameter?

MR. MESSINA: That would depend on where you are in the break spectrum, so above the transition break size you might not need to consider all of the uncertainties that you typically do today.

CHAIR BALLINGER: I mean during refuelings they do measure -- they do look at this.

MEMBER MARTIN: Okay. So they have that information. They could use realistic --

CHAIR BALLINGER: Yes.

MEMBER MARTIN: And then below -- I mean, again think about how these methods are done, the best estimate plus uncertainty methods. Pretty much everyone does non-parametric methods, create distributions, they sample from them, and you get what you get. Can we create a new -- I mean, is this a new figure of merit? Anyway, it seems a secondary to ultimately what you're looking for is just the integrity of the fuel. If you have explicitly incorporated uncertainties associated with crud and oxide into your inputs, which is -- again, it's input-driven. You get an output on the temperature of the fuel. Have you satisfied the criteria? Are you looking for something -- by explicitly saying it, are you looking for another add-on to the analysis?

MR. MESSINA: No, it's just making sure that it's considered. And this dates back I believe to a PRM that was submitted.

James, do you mind jumping in?

MR. CORSON: Yes, I was just going to -- like you said, Joey, this comes from the 50.46c days. There was a petition for rulemaking that was lumped in with 50.46c that asked us to explicitly consider the thermal effects of crud and oxide layers. So this is just something that -- when we combined basically what was in 50.46a with 50.46c for this current rule, this is something that was just carried over and not changed.

So it's just to make clear in the analysis that needs -- this needs to be accounted for somehow. So how you actually would account for it -- whether you do a deterministic thing or a statistical thing I think comes down to the licensee and whatever methods that they choose to use both above and below the transition break size.

MEMBER MARTIN: Well, and there was always this challenge because if you do an analysis to determine the maximum oxidation, right, you would start off with (audio interference), right? You'd start off with a clean rod. But if you put the oxide on there, you've already reacted, right? So either you deterministically treat it like we used to do back -- way back when and do multiple analyses -- although when you bring in the uncertainty analysis methods, you just say that, well, as long as you considered it as an uncertainty parameter, it's there as an input, it satisfies the intent of the regulation. So it's a different way of looking at the problem, but it's -- I think, Joey, you said it, that it's kind of as long as you incorporate into the input and --

MR. MESSINA: Yes, I mean, it's not --

MEMBER MARTIN: I mean, the devil's in the detail when you look at the methodologies, but you elevate it when you put of course a line item like that. (Audio interference) the crud's kind of (audio interference). Just treat the same way as he oxide?

MR. LEHNING: Sorry, Joey. Just a quick

comment on the combustible gas on that question. There are requirements for a mixed atmosphere in 10 CFR 50.44 and the guidance there. So we do -- in this regulation I think we are focused a lot on the total amount. And then there are other regulations and requirements that speak to that, just to answer that previous question.

MR. MESSINA: Next slide, please? So on fuel coolability -- so you'll notice that it doesn't -- coolability doesn't -- the phrasing doesn't change drastically from the current rule, but we had a discussion in the FRN or -- the Statements of Consideration or if you want to call it the preamble as it's now called -- its clarification on the interpretation of coolability. So we say above the transition break size. We can envision that some amount of dispersed fuel would be expected to remain coolable and safe during a LOCA so that it -- if it is shown to be safe, then it may be acceptable for LOCAs greater than the transition break size.

And but we did outline two scenarios that would remain undesirable, which is widespread brittle failure -- you don't want to fracture a significant portion of your core and lead to complete destruction of the core, core geometry, the fuel bundle geometry.

And we still want to make sure that there's no fuel or cladding melt during a LOCA. We are not saying that this is a severe accident. We're saying it's a beyond-design-basis accident. Severe accidents usually would allow for fuel cladding melt.

And DG-1434 explicitly provides guidance for analyzing the consequences of fuel dispersal.

Next slide, please? This slide just basically says that NRC-approved fuel designs that have specific ECCS performance requirements that have been established should be used in the reactor and mentions the importance of LTAs at the end of the first paragraph. And this comes directly from 50.46c, that direct final rule language.

Next slide? So I'll go into paragraph H. And this I'll note at -- this will be talked about more in the guidance that Kristy Bucholtz will be presenting after me. And this is for guidance on -- this is rule language on changes enabled by 50.46, the changes made under this rule.

An overview of this -- and I'll provide some words on each of these paragraphs. (h)(1) is about changes enabled under this section that are allowed without prior NRC approval. (h)(2) are changes enabled by this section not permitted under (h)(1), so not allowed without prior NRC approval. (h)(3) are criteria that all changes enabled implemented under this section must meet paragraph (h)(4), PRA requirements; and (h)(5), non-PRA requirements.

Next slide, please? So changes enabled under 50.46a are allowed without prior NRC approval if it's allowed under 50.59 and the appropriate -- the NRC-approved risk-informed evaluation process that demonstrates that any increases in estimated risk are minimal, and the requirements in (h)(3) are the requirements that all must changes must be -- are met, and that's no significant increase in LOCA frequencies.

Next slide? Paragraph (h)(2) is the things not permitted under (h)(1), not permitted with prior NRC approval and they must provide this list of information. And the second bullet I'll highlight. Demonstration from the risk-informed evaluation process that the total increases in CDF and LERF are very small and the overall risk remains small. And now these metrics come from the Commission SRM in 2007, SRM-SECY-07-0082 as denoted at the bottom of the slide where the Commission directed the staff that we should restrict changes to the plant to very small risk increases. And I have what is enriched numbers there.

MEMBER PETTI: So, Joe, just a question. I mean, I understand the words, but large-break LOCAs have never really dominated the CDF or the LERF in a PRA. So it would seem like there's a lot there that will not translate through in the end. Is that a fair --

MR. MESSINA: Yes, and this is to ensure that -- right now they don't dominate the CDF, but changes made under this rule -- maybe they become more significant. And then --

MEMBER ROBERTS: But, Dave, not to go back, but the whole discussion about containment pressure, the change allows the containment pressure be reduced in maybe more severe accident scenarios that are of interest and they (audio interference) because of the transition break size, because of the reduced containment pressure that goes with that. And the containment pressure for severe accidents won't go down like the loss of coolant accident pressure does. Again, just a thought of maybe that the side effects of this change might be a bigger deal from a risk perspective than the actual main change being pursued.

MEMBER PETTI: Yes, that's what I thought, but there seems to be -- I guess I'm just not convinced that all the Is are dotted and Ts are crossed in terms of -- someone ought to be running a calculation given the new rules of the road on what it really means in some of these areas. There may be secondary effects that we haven't thought of.

MR. MESSINA: Yes, and we would review these changes and -- if the ECCS evaluation determined they meet the LOCA criteria on a plant-specific basis and in the LARs and whatnot.

And then these others are that overall plant risk should not change drastically and that the -- again, no significant increase in LOCA frequencies.

Next slide, please? So this is all changes made under this section must meet the following criteria. So if it's a risk-informed rule, a risk-informed -- so we looked at some the -- these are fundamental to risk-informed regulation and that adequate defense-in-depth is maintained, safety margins are retained, and adequate performance measurement programs are implemented.

Next slide, please? So here are the PRA requirements. So they must address initiating events from sources both internal and external to the plant and that -- and for this -- I wanted to mention on the first one we would expect that low-power and shutdown conditions would be addressed with non-PRA methods. The original 50.46a rule had explicitly stated that low-power and shutdown PRAs need to be there and -- but we have relaxed that to allow for it to be addressed with non-PRA methods.

And No. 2, it must reasonably represent the current configuration of the plant. It must have sufficient technical acceptability at the level of detail to provide confidence that the total risk estimates and changes in total risk estimates adequately reflect the plant and the effect of the proposed changes on risk, and the PRAs must be -- is done under peer review.

Next slide, please? Non-PRA requirements. This is basically just saying we must reasonably reflect the current plant configuration and operating practices.

Yes?

MEMBER PETTI: Just a question. I kind of have lost a thread. I thought only Part 52 required PRAs. And Part 50 out of the alignment we were going to require 50 to have PRAs. But that's not final yet, is it?

MR. MESSINA: I'm not sure about that

rulemaking, but on this rule as a risk-informed rule we -- it's a major --

(Simultaneous speaking.)

MEMBER PETTI: So you're imposing a PRA requirement if you take this path?

MR. MESSINA: Yes.

MEMBER PETTI: Okay.

MS. CLARK: This is Theresa Clark. Just to add on. So this is one of the things that we're looking at as it goes through concurrence. The Part 50/52 rulemaking, which was considered separately, was about new reactor design applications and sort of harmonizing the requirements, whether you come in under a Part 50 construction permits space or a Part 50 combined license space.

This is sort of a different context. For this rulemaking the idea is that if licensees voluntarily want to use this alternative approach, they would largely (audio interference) PRA option. They'd largely be using PRAs for it just like they use to adopt other voluntary things like risk-informed tech specs. So we have some sort of high-level requirements for the quality of the PRA that they would need to use, but most of it's in guidance rather than in the rule itself.

MEMBER PETTI: Okay. No, thanks. That helps.

MR. MESSINA: And Kristy had her hand up.

MS. BUCHOLTZ: Yes, this is Kristy Bucholtz. So I am representing a PRA oversight branch in Division of Risk Assessment working this rulemaking. So I will talk about this more in the next presentation that I'm going to give, but the way the rule is written is it's written to work with both a PRA and non-PRA requirements, right? So there is -- the way it's written doesn't technically require you to have a PRA, but what it does do is requires you to be able to quantify both your core damage frequency and your large early release frequency in order to pass the acceptance criteria, right? So --

MEMBER PETTI: Okay. I'm not a risk expert, but --

MS. BUCHOLTZ: Right. So --

MEMBER PETTI: -- it's not (audio interference) PRA.

(Simultaneous speaking.)

MS. BUCHOLTZ: Right. So I'll talk about it more, but if you can quantify it without using a PRA, right, you can use it. In other words, if you're making a change that can be qualitatively assessed and you have your baseline numbers already established -- because most of our operating fleet right now has already adopted the Surveillance Frequency Control Program.

So we all -- they all -- all the current operating light water reactors have a -- at least a base-level PRA available. So they could use that model that they already have and their numbers in -- they can use that model along with new qualitative non-PRA assessment methods to use this rule, if they choose to. And I'll talk about it more in the next -- in the risk-informed discussion that I'll be giving next, so --

MEMBER PETTI: Okay. No, thanks. That helps.

MR. MESSINA: Thanks, Kristy.

Next slide, please? So this paragraph states that the director of NRR can impose restrictions on operation if ECCS cooling performance is not consistent with the requirements. And this is maintained from the current rule. And every proposed iteration of the rule has had this as well, so this isn't anything new.

Next slide, please? On reporting, (j)(1) and (j)(2) talk about ECCS reporting and corrective actions. It's largely the same as in the current rule just with some clarifications on reporting requirements and it matches largely what was in 50.46c except for this first sub-bullet where we eliminated reporting requirements for changes or errors that do not result in an inability to assure compliance with 50.46a until an SDA or a DC is referenced in an application for a CP, OL, COL, or manufacturing license. So there's no impact on safety if no one's currently planning to build a plant with this SDA or DC. So we didn't think that it would -- they would need to report these actions as the operating fleet would. And this parallels what was proposed in the Part 50/52 Alignment rulemaking.

(j)(3) is on risk assessment reporting. And I mentioned earlier that the five-year PRA updates were removed from the rule during concurrence and concurrently this risk assessment reporting was removed from the draft proposed rule during concurrence.

Next slide, please? So under (h)(1) it allows minimal changes without prior NRC approval. Any minimal changes made under that paragraph would need to be submitted every 24 months and there would need to be a basis for the changes not invalidating the transition break size.

Lastly, the welding inspection report must be submitted within 120 days after completing an outage where the inspections were performed. And this can be combined with the summary report required under 50.55a.

Next slide, please? On paragraph K, first we have what's considered as significant change to an ECCS evaluation model for LOCAs at or below the transition break size, and we describe it for traditional UO2 zirc fuel. And that would be as is today PCT greater than 50 degrees as well as we added oxidation greater than 1 percent. And this matches what was in the draft final 50.46c rule.

Next slide, please? For LOCAs above the transition break size a significant change or error could be proposed different from the previous paragraph, but if alternate criteria are not defined, then the same reporting criteria would be expected to be applied to 50 degrees PCT and 1 percent ECR. And then paragraph 3 allows for non-UO2 zirc fuel to -- they would have to define their own -- what is significant change.

Next slide, please? Go back. Okay. So lastly, I wanted to discuss the changes to the base

50.46. So there were some corresponding changes we made to 50.46, mainly in applicability. 50.46 or 50.46a could be used if XYZ, which kind of parallels the applicability section in 50.46a. We did not want to touch much of the other criteria in 50.46 such as the 50.46b criteria, such as 2200 PCT and 17 percent ECR because we wanted to as much as possible limit the scope of the rule and possible delays that would be entailed by changing these requirements.

Our rule is voluntary, so there are other considerations that we would need to go through if we decided to change those 50.46b limits. But we did add a statement that if you do not meet the cladding and fuel specified at the beginning of current 50.46, which says UO2 pellets within zircaloy or ZIRLO cladding, that you can meet -- instead you could choose to meet 50.46b, so the current rule, 2200 or 17 percent, or if you meet 50.46a(f)(1), which is the address cladding degradation and those four criteria that I went through previously. So this allows for new fuel types to be implemented without an exemption if people wanted to pursue it that way or they could elect to still request an exemption to 50.46b criteria.

Walt?

MEMBER KIRCHNER: Yes, thank you. Cladding degradation is rather an amorphous phrase. I want to point back to this 17 percent. The 17 percent; and perhaps I could be corrected if I'm wrong, is different than the 1 percent of the zirc being oxidized. That was more with regard to hydrogen generation and concerns about combustion either in the reactor system, primary system and/or containment. But the 17 percent, if I remember correctly, really had its origin in the idea that that was a limit that would to first order provide structural integrity for the zircaloy claddings.

MR. MESSINA: Yes, the 17 percent was --

MEMBER KIRCHNER: So that core coolability would be maintained to first order and configuration of the core would basically remain intact. So you're opening the door to essentially large-scale core relocation if there's not further specificity on what you mean by cladding degradation.

Now the end state, as has been pointed out by one of my colleagues in a previous subcommittee meeting -- I mean, the TMI core melted down substantially but was coolable. That's probably not what we're looking for in a design-basis accident situation in an LWR. So where are the boundaries on this and how much relocation and reconfiguration of the core do you see being acceptable?

MR. MESSINA: So yes, the 17 percent is meant to preclude embrittlement of the cladding for structural purposes and maintaining -- keeping the fuel pellets in the cladding. To address cladding degradation phenomena, that -- they would be expected to address that oxidation, so the oxygen ingress into the cladding that effects -- the embrittlement effects of that. That is just in guidance.

So there's a 50.46c graph that says percent ECR is a function of hydrogen. So that is in DG-1263 that James will present tomorrow. So we would still expect licensees to address this cladding embrittlement that is addressed by the 17 percent. The 50.46c research findings stated -- basically found that 17 percent was not necessarily conservative with the 2000 PCT limit due to the fact -- due to the impact of -- the impact that hydrogen has on embrittlement of the cladding. So that curve of ECR versus hydrogen is in that draft guide and we would expect them to address that embrittlement mechanism.

MEMBER KIRCHNER: But just for rhetorical purposes to explore the implications, what's your sense of what would be acceptable in terms of a large relocation of the core?

MR. MESSINA: We don't necessarily know. If there is a large relocation of the core, I would expect that licensees would be -- would provide that basis. James has a Reg Guide addressing the consequences of fuel dispersal, so that's kind of how they would demonstrate safety

James, go ahead.

MR. CORSON: Yes, this is James Corson from the staff. So yes, as Joey said, I'll be talking about this more tomorrow when I go into Draft Guide 1434. So that's just like one -- that Guide basically represents like one idea of what we would find acceptable. So at a high level what we're trying to do is prevent melting of the fuel or some wide-scale brittle failure of the cladding. And so like the existing 50.46 criteria were really meant to prevent that latter scenario where you have wide-scale brittle failure.

MEMBER KIRCHNER: Exactly.

MR. CORSON: And we still have requirements for that. On top of that, this new requirement where you have fuel dispersal, we also want to prevent that from escalating into a situation where you either melt fuel or that bumps up your cladding temperature enough to lead to widespread brittle failure.

So that's what -- the Guides are trying to prevent those sorts of situations from happening and they do lay out ways that we find acceptable to meet these requirements.

MEMBER KIRCHNER: Yes, the guidance is not regulation.

MR. CORSON: It is not. But yes, I mean, I think in a lot of ways we're moving in a way to make our regulations a little more flexible and to have the details in guidance.

MEMBER KIRCHNER: So you want to allow the flexibility to have a widespread disruption of the core as a design-basis accident?

MR. CORSON: So I think the guidance is meant to prevent a wide-scale disruption. What we're talking about is a much -- so if you're talking about fuel dispersal due to ballooning and burst of the cladding -- I'll get into this a little bit more tomorrow -- but so far analyses suggest it's a pretty limited amount of fuel. So certainly less than 1 percent of the fuel. Perhaps much less than 1 percent of the fuel. So it's relatively limited. And what we're trying to do is prevent that from turning into widespread disruption of the core. So yes, we don't want widespread disruption and that's what our requirements are trying to prevent.

MEMBER KIRCHNER: I'm not seeing it. And remember, this is just one member's opinion. Now I'll wait to see -- I will be interested to see the guidance, but the implications here are significant.

Now in turns out if we're being candid with each other, we know with the existing fleet and estimates of higher burnup with some of the advanced zircaloy claddings that at breaks below 10 or 12 inches, whatever, in a PWR the differential pressure is probably not going to lead in a -- in one of these LOCAs to ballooning and discharge or dispersal of fuel. So I'm trying to explore why we're not asking the licensee to preclude this situation.

In other words, one criterion that we've seen some applicants use is no rupture. That takes the dispersal off the table.

MR. MESSINA: Yes, we think that it's possible for industry to demonstrate that some amount of fuel dispersal into the coolant would be safe, and we'll see -- we provide guidance on that.

And on the other topic of moving some of this into guidance, we do have that 50.46c SRM from earlier this year that explicitly asked us to consider whether certain prescriptive criteria such as core temperature like the PCT should be placed in -- should be codified or placed into the guidance. So part of this is based on the Commission SRM that we basically need to have -- be more performance-based in our regulations and maybe have more specifics in guidance.

MEMBER KIRCHNER: No, I accept that. I mean, there's nothing magic about 2200 degrees Fahrenheit. It's highly likely that there would be wide-scale ballooning if a large part of the core reached that temperature, and probably well before 2200. So that's a historical number. It was, at the time in the '70s, thought to be a rather conservative estimate of structural integrity for zircaloy clad.

We've seen quite a bit of progress both in -- on behalf of the fuel vendors, they've improved the cladding materials significantly in terms of hydrogen uptake and other issues that make for a more resilient clad. But the direction is to preclude rupture is a much more reasonable criterion than 2200 or some magic 17 percent oxidized.

That's an observation, not a question, but I'm concerned that the -- what you're proposing for the rulemaking doesn't necessarily -- where you take out the peak clad temperature and you take out the 17 percent oxidation that you're getting a comparable set of requirements in terms of core performance under a design-basis accident.

CHAIR BALLINGER: This is Ron Ballinger. As a practical matter that's why I was asking the question this morning. The data that we now have related to embrittlement and the like, whether that would get translated into it and be laid on people that did not want to use the rule. Because 17 percent with the new data that they have doesn't really exist as an allowable because as soon as you get any burnup with any hydrogen, with any cladding, the allowable is less than 17 percent, for sure.

And so I'm still confused on how that data would not -- you say backfit would be -- somebody comes in and says I want to use 17 percent and you come back and say, well, wait a minute, this data that we have here says you can't. Well, you say, well, the rule says I can. Now what do you do? Because if you're lower than 17 percent, that in effect lowers the 2200. It doesn't lower the limit, but it -- during an accident if you're allowed only 14 percent, whatever the number is, to get to that you have to -- to get to that number you'd have to be at a temperature much longer and at a lower temperature than 2200. Am I reading this wrong?

MEMBER KIRCHNER: No, I'm not trying to defend or question the 17 percent. The functional intent of the 17 percent was to maintain cladding integrity so that the core didn't turn into a rubble bed. So I don't see substituting cladding degradation as achieving the same intent. How much degradation?

MR. MESSINA: That's in guidance. So basically the 17 percent. Not exactly 17 percent, as Ron talked about. The oxidation as a function of hydrogen has been placed in the guidance to achieve that objective.

CHAIR BALLINGER: But isn't the oxidation limit, regardless of what it is, designed to prevent brittle failure of the fuel?

MR. MESSINA: Yes.

CHAIR BALLINGER: Yes.

MEMBER KIRCHNER: Yes, that's what it was there for.

CHAIR BALLINGER: Yes.

MEMBER KIRCHNER: Especially when you quench the core during re-flood.

MR. MESSINA: Yes, maintain post-quench ductility, yes.

CHAIR BALLINGER: Yes. Tomorrow will be interesting.

(Laughter.)

MR. MESSINA: Yes, we believe that --

MEMBER PETTI: But philosophically this is -- performance-based regulations allows flexibility. And this is the trade-off that when you try to pull back and do something that's a little bit higher level, you don't get that specificity, which I guess gives you a warm fuzzy. This gives you the flexibility. That's the trade-off, right, to the -- we've discussed this in Part 53 ad nauseam, as I remember.

CHAIR BALLINGER: But I don't see that the guidance gives you a loophole. There's no loophole here where all of a sudden you can go some other direction.

MEMBER PETTI: No, that's --

MEMBER KIRCHNER: My point, Dave, isn't -- I'm not defending 17 percent or 2200. I think I already made a case for no burst criterion, but the -- what I'm not seeing yet, and maybe it will be in the guidance, is a little more clarity of intent for the regulation itself. Degradation is rather a vague phrase. Seventeen percent is rather an arbitrary limit. So I'm not going either way. The intent though, that's what I'm not seeing. The intent was to maintain the core integrity. And can you state that in a performance manner rather than a prescriptive temperature or oxidation limit?

DR. BLEY: Walt, this is Dennis Bley. This is pretty naive, but it would seem that the rule could state some high-level -- as you were saying, some high-level statement of what --

MEMBER KIRCHNER: Intent or objective. Objective is a better word.

DR. BLEY: Yes, I mean you're trying to avoid wide-scale, kind of a definition of wide-scale and we don't want this to happen. Then the guidance tells you a way to meet that intent. But with the intent not being in the rule it leaves it pretty much up to who's ever around at the time deciding what it means.

MEMBER KIRCHNER: Yes.

MR. MESSINA: Yes, and I think some of this will probably be answered tomorrow in the Reg Guide. And on the rule we do have that -- one of those high-level criteria is maintain fuel coolability. And as Ron has said, the PCT and 17 percent has traditionally been used to ensure core coolability in the past. So that overall objective is still in the rule, maybe just not as explicit as --

MEMBER KIRCHNER: But again, you can cool a debris bed, but I hardly consider a debris bed in the bottom of the reactor vessel as a design-basis accident, acceptable design-basis accident.

MR. MESSINA: Well, we are saying that LOCAs above transition break size where we would allow for this fuel dispersal is a beyond-design-basis accident.

MEMBER MARTIN: This line of questioning seems to be what exactly are the analyses above the TBS intended to protect? Severe accidents tend to focus on anything but the fuel, right? You're -- looked at the defense-in-depth analysis. You are -- it could be Chapter 19. The loads in severe accidents are used in Chapter 3 for structure, containment structure, non-safety equipment used to mitigate those events. Maybe I&C could be most safety and non-safety.

The ways things are looking here is that you would expect a limited scope of equipment to depend on analyses, the break size of EDS, and probably not the fuel. If you eliminate -- you make things best estimate, you eliminate single failure.

Those analyses will probably be -- show lower loads than your limiting rate below TBS. I don't think we got clarity on what exactly this -- these analyses above that size really cover.

Now, we can think about it and maybe come up with our own list, and maybe in some of the guidance you -- it's being clarified. But I think that's kind of questioning we're getting into. And it kind of goes back to the containment discussion. But I would not think you would redo analyses of break sizes above TBS for every reload because it would not expect that a beyond-design-basis accidents analysis would be looked at every single time. It should be very easy to disposition just like the severe accidents in Chapter 19 are easy to disposition.

But anyway, I think we're all going to go back and look through and try to get some clarity on the -- how do you delineate what is used where when it comes to using analysis to determine adequate protection. But I expect anything beyond-design-basis is really kind of a defense-in-depth kind of analysis.

More a comment, but the question is out there is that 100 percent sure whether those analyses being used for to demonstrate a protection. And why would you do -- necessarily do a beyond-design-basis accident analysis for every reload? It's only if you're looking at it to assess the fuel, the integrity of fuel, but then that becomes design-basis for the fuel. Make it circular.

CHAIR BALLINGER: Any other questions?

(No audible response.)

CHAIR BALLINGER: I think tomorrow will be an interesting -- like I said, an interesting discussion tomorrow. And I'll have to go re-read the preamble because I think there's something in there in this -- what you call -- we used to call the Statement of Conditions.

All right. So it's 3:49. I propose that we take a break until 5 after 4:00. That' would be 15 minutes. So until that, we're in recess until 5 after 4:00.

(Whereupon, the above-entitled matter went off the record at 3:49 p.m. and resumed at 4:05 p.m.)

CHAIR BALLINGER: Okay, we're back in session and Kristy is remote, and so the floor is yours.

MS. BUCHOLTZ: Thank you, and good afternoon to everyone there. Like you said, my name is Kristy Bucholtz and I am representing the PRA Oversight Branch, which is in NRR's Division of Risk

Assessment.

My presentation today is on Draft Guide 1426, and I managed to make it the longest title possible and maybe next time I will consider cutting it down a little, but, so it is an approach for the risk-informed evaluation process and it's to support the alternative acceptance criteria for the emergency core cooling systems for light water reactors.

As part of the development of this proposed rule, the NRC staff included the risk-informed evaluation and the risk-informed evaluation process that were included in the previously proposed by discontinued rulemaking on the risk-informed changes to the LOCA technical requirements. This draft guide provides implementing guidance for that process and addresses previous Commission direction.

As part of the development of the previously proposed rule, the staff had received comments. These comments were reviewed and changes were made to the previous rule as a base point and this draft guide.

So, going forward, the staff will consider any additional considerations or feedback that is provided on the risk-informed evaluation and the risk-informed evaluation process that we receive during this rule-making process. Next slide, please?

So, I want to start off by talking about the approach in the rule. So, paragraph (c)(iii) requires a risk-informed evaluation to make any proposed change. The way the rule is established, it has basically two risk pathways. The first pathway is on this slide and the second pathway is on the next slide.

The first pathway is you can individual, or individual submittals can be made for this application. It would require a risk-informed evaluation which would include a risk assessment of the proposed changes.

It would allow multiple submittals. It would be set up so that you have an initial adoption of the rule, and then every future change enabled by this rule would be also then submitted, right.

So, the initial and each enabled change would be submitted for NRC's review, and then the requirements that would need to be met are those requirements that are in paragraph (h)(2). And it is the equivalent of the acceptance guidelines in the Draft Guide 1426, and I'll talk about those risk requirements as we get further into the presentation.

So, the first pathway that an entity can do would be to do, well, like I said, I consider to be the individual, somewhat independent changes, right. Each one of them would be submitted and the risk requirements in (h)(2) are the only requirements that will apply. Next slide, please?

So, the other pathway that's in the rule is that you can -- it's a voluntary process that they can -- the whole rule is voluntary, but then there's a voluntary process inside the rule where an entity can decide to submit information for a risk-informed evaluation process.

So, it is, that is still done through a submittal to the NRC, all right. It still contains a risk-informed evaluation with the risk assessment of the proposed changes. What differs here is that the risk-informed evaluation process is also submitted to the NRC for approval.

So, they would provide a risk evaluation that goes through this rule and shows that it's appropriate to adopt the rule, and then as part of that submittal, it would lay out a process that would then be used to review any future enabled changes, and then it gets screened slightly differently.

So, this process also is done through multiple submittals, or can be done through multiple submittals, and it also requires an initial adoption of the rule, right, but that initial adoption, that license amendment that comes in is going to have the risk-informed process laid out in it in addition to the risk-informed evaluation, right, like I said, and that process is then used for future changes that are enabled by this rule.

So, as an entity would review the future changes, they would use their process, and it's envisioned that they would use the process and come up with a calculated risk criteria, right. We've included this in the rule as (h)(1) and (h)(2).

If they calculate the risk of the changes that they want to make and they are low enough that they meet those provided in paragraph (h)(1), then they can make the, a change without NRC approval. So, it will use 50.59, and we'll talk about that in later slides, but they will be able to make the change using 50.59 without NRC's approval.

However, if when they calculate the risk, if the risk is higher than the threshold that's in paragraph (h)(1), but it's lower than that in paragraph (h)(2), they can still make the change. The difference is it would then be submitted to the NRC for our review.

So, they can still -- so they have two different pathways in this pathway itself, right. So, if they choose to do the risk-informed evaluation process, they can use the two different metrics. If they meet the one, they can make the change on their own. If they don't meet the other one, they can still submit it for NRC review and approval.

If it's above paragraph (h)(2), then it would not be allowed, right. And like I said, I'll talk about the risk metrics once we get through. So, I just wanted to lay out that framework first. So, there's two pathways.

You don't have to, an entity does not have to adopt this risk-informed evaluation process that's laid out in this rule. It's just a benefit to them if they want to go ahead and choose to use it, so I wanted to make that clear first. Okay, so please go to the next slide?

So, now let's talk about the Draft Guide 1426. So, this is the structure of the draft guide. It follows the same structure as Reg Guide 1.174. It lays out the four elements that are in 1.174, right.

The first one to define the proposed change, the second one is to perform your engineering analysis, the third one is to define and implement the monitoring programs, the fourth is to submit the license amendment, and then five and six follow on with quality assurance and documentation. So, Draft Guide 1426 is structured in the exact same way as Reg Guide 1.174. So, next slide, please?

So, the first element was defining the proposed change. This element follows the same structure as Reg Guide 1.174. However, Reg Guide 1.174 is written for licensees that want to do a change to their licensing basis. So, we did take the wording that is in 1.174 and we did have to modify it a little bit.

So, we did modify it from licensee to entity so it makes it clear that it follows the rule, and then we modified it from a licensing basis change to a proposed change because some of the changes that are allowed in here could be procedural changes that are outside the licensing basis, but impact your risk assessment, right.

So, we did change it to say it's any proposed change that is being enabled by this rule, right, and then we added sentences for an entity to identify the aspects that may be affected.

So, whereas Reg Guide 1.174 just assumes that you're changing the licensing basis and it assumes you need to use 10 CFR 50.90 for a license amendment request, this program is structured where you may be able to do changes under 50.59 once you adopt the rule.

So, it does have written into this guidance that an entity would look at the change, see what it's affecting, whether it's the licensing basis, whether it's an entity-controlled documentation or anything else, and then choose the proper implementation pathway that they want to pursue, whether it's going to be under 50.90 or whether or not it might be through 50.59. All right, next slide, please?

So, for Element 2, Element 2 is performing the engineering analysis. This is the main portion of the document, and so this is kind of how it's structured. So, Section 2.1 is the risk-informed evaluation process, with Section 2.2 being all of the risk assessment requirements, or I should say guidelines, not requirements, guidelines in there.

So, Element 2 starts with an overarching guidance that the scope, the level of detail, the technical acceptability of the engineering analysis should be appropriate for the nature and the scope of the proposed change, and that uncertainties should be considered.

And regardless of the analysis method chosen, right, you still have to show that you meet the key principles. In this case, the initial adoption of this rule would be to show that your regulation is being met.

So, the four principles laid out in the reg guide are the increase in risk, your increases in your risk criteria, right, your defense in depth, your safety margins, and your performance measurement programs.

So, when it comes to Section 2.1, this section discusses that risk-informed evaluation process, and it's at a high level to start with, and that it should be used to assess each enabled change.

This section also discusses that the risk-informed evaluation process should be able to evaluate the risk assessments that are going to follow under Section 2.2, right. It also includes your performance measures, some discussion of the performance measures, which are further discussed under Element 3.

And then the risk-informed evaluation process should be able to evaluate both risk pathways, whether an entity choose to pursue 10 CFR 50.59 and if the minimal risk criteria is met or if they choose to go 10 CFR 50.90 and put in a license amendment if the very small criteria is met.

So, Section 2.2, like I said, discussed the risk assessment. The goal of this section is to estimate the total increase in risk from any facility technical specifications and procedural changes that are enabled by this rule, and it addresses the principle that the proposed increase in the core damage frequency and the large early release frequency are less than that small and are consistent with the intent of the Commission's safety policy statement.

This section discusses establishing a baseline risk, excuse me, establishing a baseline risk estimate to which comparison can be made. It allows for both quantitative and qualitative assessments, and it encourages a full-scope PRA, but doesn't require it. It allows for the exclusion of out-of-scope modes.

So, under Section 2.2.1, it discusses that the probabilistic risk assessment, this section, it immediately calls attention to Reg Guide 1.200 as one approach acceptable to the staff for the acceptability of PRA results for risk-informed activities, right.

It states that when Reg Guide 1.200 is used in support of this application, it will obviate the need for an in-depth review of the base PRA, allowing staff to focus on the key assumptions and areas identified by the peer reviews as being of concern and relevant to the application.

This is the way we do most of our license amendments that come in right now for the risk-informed completion times, right. When Reg Guide 1.200 is used for those, we focus on those peer reviews and the findings and observations that were put in by the peer areas to ensure acceptability of that, and so it is envisioned that this rule would be structured in the same way for this risk approach for the probabilistic risk assessment, right.

This section then goes on to address the PRA's scope, the level of detail, and the technical elements, and the plant representation, and the PRA configuration control.

And throughout the whole section, it continues to point to Reg Guide 1.200. It isn't trying to replace it, but it's trying to work together with Reg Guide 1.200 so we're not duplicating any more than we need to for this draft guide with our current position in Reg Guide 1.200.

So, Section 2.2.2, it discusses the non-PRA risk assessment methods. It states that the impact of the proposed change on operating modes and initiators that are out of the scope of the PRA should be evaluated and estimated, right.

So, non-PRA methods should use and produce realistic results, and should either quantify an estimate of the change in risk or it should demonstrate that the change will not significantly affect the out-of-scope modes and initiators.

So, this is where you can use a qualitative assessment to show that you don't have to do your shutdown in low power modes if they are not a mode that would be significantly affected by the change that's being proposed, right. The result --

MEMBER MARTIN: Kristy? This is Bob Martin.

MS. BUCHOLTZ: Yes?

MEMBER MARTIN: Could you elaborate a little bit more on maybe the scope of what non-PRA risk assessments? You mentioned qualitative as kind of the lead-in. I just am not sufficiently familiar with alternative methods like, you know, considered in this domain of non-PRA.

MS. BUCHOLTZ: Yeah, so because this evaluates, it evaluates all internal events as well as all external events, so like a good example here of, we'll call it a non-PRA method, would be like a seismic margin analysis.

You're not using a seismic margin, or you're not using a seismic PRA to do your calculations, right. You're using the seismic margin analysis that's a well-established method that is known throughout the industry.

So, this is written on a generic level to use non-PRA risk assessment methods. There are other methods out there. Like I said, all of them for the most part should be NRC-reviewed and approved methods, and we did put that in the guidance, but it would be a method like that that would be used.

MEMBER MARTIN: What size of margin is that? That sounds quantitative.

MS. BUCHOLTZ: Yeah, it can be.

MEMBER MARTIN: So, it's a mix?

MS. BUCHOLTZ: Yeah, it's a mix. So, it can be qualitative. It can be quantitative. It depends on what type of method is chosen whether it will quantify or not. It could also be, for a non-PRA risk assessment method, it could be a bounding assessment as well, right.

So, they could do a bounding assessment that would take like a seismic penalty or put a penalty in place for the risk that is a conservative penalty that could be done depending on, like I said, what it is that they're trying to estimate the risk on.

MEMBER MARTIN: Yeah, one thought in looking at the draft guide would be a footnote to expand on maybe a definition of non-PRA risk assessments just to clarify them.

MS. BUCHOLTZ: Okay, thank you. I will take note of that.

DR. BLEY: Kristy?

MS. BUCHOLTZ: Yes?

DR. BLEY: Dennis Bley. Since you talked about NRC-approved non-PRA methods, I'm very familiar with the seismic one you talked about. Do you have a catalog of those or examples of them that you can point us toward?

MS. BUCHOLTZ: I would honestly have to check and see if we have them written down. I mean, we have different ones that are known that are used all the time during, you know, fire assessments and for seismic, but I'm not sure if they're listed anywhere. I will actually have to check on that.

DR. BLEY: Yeah, it seems there has to be at least some stemming quantitative aspect to it if you're going to really make claims about risk from it.

MS. BUCHOLTZ: Right.

DR. BLEY: And the margins' analysis is partially, at least the ones I'm familiar with, are at least partially quantitative. They just don't carry out the full risk assessment. Okay, it would be good to see that.

MS. BUCHOLTZ: Okay. Okay, for Section 2.2.2, the non-PRA risk assessments, the resultant, your resultant total change in risk should be tracked and then combined with any PRA assessments.

So, if you do, if an entity does a mixed blend of a risk assessment where they're using both a PRA model to do the in-scope changes and they use a non-PRA method for out-of-scope changes, that any risk that's calculated from the non-PRA should be added to the PRA assessment together for the total risk.

And it allows, and the non-PRA risk assessment also allows use of maintenance rule to determine risk significance if your equipment is scoped into the maintenance rule, so that can be -- you can use that as a -- assuming that, like I said, assuming you can show why it's technically adequate, and that's the last part, right.

So, any basis for why the method employed is technically adequate for the application should be documented. So, if you use maintenance rule, the seismic margin, or any other non-PRA risk method, it's expected that the entity would just document why it's adequate for this application to be used.

CHAIR BALLINGER: Vicki?

MEMBER BIER: Quick question. You said any non-PRA assessments would be added to the PRA results, and I assume that does not mean like quantitatively added, like summed, but combined with or presented together with, is that correct?

MS. BUCHOLTZ: Well, it's both. So, if it was a qualitative assessment, it would be used to show that it doesn't need to be -- so they would use it to show that whatever they're analyzing is not a significant contributor to the risk.

MEMBER BIER: Yeah.

MS. BUCHOLTZ: So, in that case, it would be used in one package together to show that you've covered all of your bullets. However, if they were using a non-PRA risk method that did quantify a number, that number would then be added to the PRA assessment, and that's assuming that you're not double-counting, right. So, we wouldn't --

MEMBER BIER: Yeah.

MS. BUCHOLTZ: -- assume that they're using their PRA to do, you know --

MEMBER BIER: Sure.

MS. BUCHOLTZ: -- let's say they have a seismic PRA, then they decided to do seismic margin separately, and then, you know. So, we wouldn't want double-counting, so the thought is that --

(Simultaneous speaking.)

MEMBER BIER: I guess it's --

MS. BUCHOLTZ: Sorry.

MEMBER BIER: It's difficult for me to visualize how a non-PRA method would give a probability as an outcome that could be added to the PRA probability, but maybe I'm just not aware of what people are doing.

MS. BUCHOLTZ: Yeah, this comes in a lot of times when they'll put like a penalty in place, like sometimes they'll put a penalty in place saying hey, we're not going to analyze this because they don't want to go in-depth and they can live with the penalty that they establish, right.

So, if it's for, let's say, an external hazard, and they have a penalty that they want to put in place, when they do that, usually a lot of times it's a quantitative number that they are saying hey, for, you know, like high winds, we're just going to assume this number, right.

MEMBER BIER: Okay.

MS. BUCHOLTZ: So, it gets --

MEMBER BIER: So, I guess --

MS. BUCHOLTZ: -- done like that.

MEMBER BIER: -- I would describe that as a bounding analysis or something, is that fair?

MS. BUCHOLTZ: Yeah, bounding analysis --

MEMBER BIER: Okay.

MS. BUCHOLTZ: -- would also fall to be a non-PRA risk method --

MEMBER BIER: Okay, got it.

MS. BUCHOLTZ: -- because you're not analyzing it.

MEMBER BIER: Now I understand what --

(Simultaneous speaking.)

MEMBER BIER: Thank you.

MS. BUCHOLTZ: Yeah, yeah. Okay, so Section 2.2.3.3. discusses the risk metrics contained in the rule and states the increase in core damage frequency and the large early release frequencies should be evaluated and compared to the acceptance guidelines.

And Sections 2.2.3.1 and 3.2 will be

talked about in a moment. So, I have those on the next slide because they say a picture is worth a thousand words, so I put it in a picture, all right.

So, Section 2.2.4 discusses defense in depth, right. This section summarizes Reg Guide 1.174 and it points to Reg Guide 1.174 for the full discussion. It's consistent with Reg Guide 1.174's intent and it states that entities should retain a level of mitigation to ensure that mitigation capabilities are maintained for accident sequences that lead to relatively late containment failure and result in late radiological releases to the public.

So, because of the transition break size here and the fact that we're moving everything into beyond design basis for everything above that transition break size, right, we want to evaluate to make sure we're not changing anything. So, I mean, we're concerned about large early release frequencies, but we also want to make sure that we're not going to cause anything on the late end.

So, we do have it as a reminder to entities that, depending on what proposed change you're making, that they should verify or look at containment failures on the late side and make sure that they are not going to result in a late radiological release to the public, so we're still concerned with those.

And then Section 2.2.5 discusses safety margins, and that section is the exact same as Reg Guide 1.174. Like I said, we look at the modifications to discuss proposed change in entities, so it is the same. All right, so next slide, please?

All right, so Element 2, Section 2.2.3.1, which is your acceptance guidelines for the risk-informed evaluations that require prior NRC approval, so I want to, before I go any further on this slide, I just want to say that this is a picture that is not being used anywhere other than in this presentation.

We are not rolling this into any guidance, right. This is just to help visualize where we are for the acceptance criteria, so I just want to clarify that first.

So, for the risk metrics that require NRC approval, paragraph (h)(2)(ii) in the regulation requires that your total increases in core damage frequency and large early release frequency to be very small and that the overall risk remains small.

So, it uses and points to the figures in Reg Guide 1.174, Figures 4 and 5, which are shown on the screen minus the red edits, right, so this is pictures of those figures marked up for where we're at.

So, in the application here, the result in increases above your core damage frequency, above ten to the minus sixth per reactor year, or the large early release frequency above ten to the minus seventh, will not be considered, right.

So, on the left side there, you have your change in core damage frequency and your change on large early release frequencies. So, we're talking for the change going up the side, right, which is why we show the X-ed off region.

So, normally in Reg Guide 1.174, you would be able to make changes if you fell into region two, which is considered the small region, and in this case, we're saying that for the very small changes, you can't be in that region at all. You need to be in the region three, which is underneath it, right, so below less than the ten to the minus six, right, and then you also need to meet the small criteria for your baseline.

So, we're pointing down that the small criteria is ten to the minus four for your core damage frequency, and large early release frequency is ten to the minus five. And I will stop there and take questions.

DR. BLEY: Dennis Bley again. What's the rationale for having a different guideline here than we have in 1.174?

MS. BUCHOLTZ: So --

DR. BLEY: This is for a change, right, yeah.

MS. BUCHOLTZ: Right, so there's two rationales. So, and I do want to say I was not around in 2010 when we did this, or at least I was not working in this area for that, so I was not involved, so I don't know the full discussions of what happened.

But what I can find is that in that little, the box underneath, which I know is hard to read, I've got it on there just to kind of show what it is, but out of the Commission's direction and the staff requirements in the SRM SECY 782, right, the Commission basically said in order to more closely follow the approach presented in Reg Guide 1.174, that the staff should modify the proposed rule to ensure that any changes under this rule be further restricted to very small increases, notwithstanding the fact that they would otherwise be permitted under 50.59, right.

So, we do have Commission direction right now that has not changed since that time frame when this rule was previously looked at in, like I said, the 2008-2007 time frame, so that's one reason that we went with -- so that's one reason that it is very small, to stay consistent with that, but then the other --

DR. BLEY: That reason kind of locks you in.

MS. BUCHOLTZ: It does lock us in, it definitely does, but the other reason too, and so, because we were, or I was not around then and I don't know exactly what changes the industry had proposed at the time, and so when I look at this rule, if you're not making --

So, if an entity came in and they just wanted to adopt the transition break size right now, let's say they weren't making any changes other than they want to move it out and it's to use new fuel, right, and they want the enrichment, so they move the transition break out of the design basis accident into that beyond design basis space. If they make no other changes than that, most of your PRA, right, will not be affected. There will be no change in your risk there.

So, the transition break size frequency falls at that 1e to the minus five range, and so if you were less, if we allowed the small changes area, it basically wouldn't be risk informed. It would only look risk informed because your TBS frequency is 1e to the minus five, and any change would be so small, right, that it would basically allow everything to go through.

So, it's not until a licensee starts to make other, or an entity starts to make a lot of other changes in, you know, systems, structures, components, other things, that it would actually start to screen. So, part of the thought was it's on the lower end. If an entity comes in and doesn't make a lot of changes right off the bat, it would allow everything and it wouldn't really screen much.

DR. BLEY: Yeah, well, maybe that's okay, but I -- you can't do anything given the Commission's guidance, but I don't understand the basis for it.

MS. BUCHOLTZ: Yeah, yeah, I agree, and that's why I'm pointing it out. So, the Commission did give us that direction so far, so we did keep it consistent with that, so, and until we have further information that might give us better ways of it being used to show that it's overly restrictive, right now I don't see that as the case until I get some more information that would let me know that that is too restrictive, so.

CHAIR BALLINGER: Walt?

MEMBER KIRCHNER: Yeah, in context of the earlier presentations, how are you defining core damage?

MS. BUCHOLTZ: We have not changed it. So, for this, we have not changed the definition of core damage, right, so it is still severe core-wide damage that occurs.

So, FFRD in this case is not playing the part here because this is severe accident, so we haven't changed that part, at least currently not. We've given some consideration to could we do something along that area, but we have not decided to push through that at this exact moment.

MEMBER KIRCHNER: So, I'm just trying to play this out mentally then. How much dispersal and relocation of fuel would be allowed and still not have core damage?

MS. BUCHOLTZ: Yeah, that one I can't answer. I would need James and them to help me with --

MEMBER KIRCHNER: Then how do I use this?

MS. BUCHOLTZ: Well, this is --

MEMBER KIRCHNER: I'm confused. I'm

really confused. And what does delta CDF mean?

MS. BUCHOLTZ: Okay, I got you for sure.

MEMBER KIRCHNER: Because you have a sliding scale.

MS. BUCHOLTZ: Right, yeah, you're still, so you're still -- this -- so there's two different things going on in this rule. So, they're talking FFRD, but when we are talking about using this for that, you're still analyzing the, you're still analyzing all of the breaks greater than your transition break size. You're still doing that to the PRA and to a full melt, right.

So, you're still looking at a full core-wide damage. That's not changing. So, the PRA analysis isn't evaluating the TBS from the FFRD standpoint. That part is being done under the ECCS and the TBS evaluations. So, this is still analyzing -- this part of the rule is still analyzing the breaks that are being made from any changes from the structure, systems, components failure-wise.

So, if you decide to, upon doing a realistic estimate of TBS, you decide to remove, you know, what you're crediting for ECCS systems, you could restructure your, a licensee, if they wanted to go further in-depth, could restructure some of their event trees and stuff based on that, although we don't expect them to remove, you know, ECCS components and systems because of this rule.

But if they were to go along those lines or to change, you know, like change diesel requirements or change other things that can be evaluated, we would still do that using the failure criteria, and the standard core damage frequencies, and the risks calculated from the core-wide damage. We wouldn't be analyzing the FFRD directly. That would be done under the other portion for the ECCS requirement.

Make sense, or at least do you guys understand or do you have further questions on that? So, this is evaluating the risks to the plant and to the structure changes you're going to make to tech specs and stuff like that still using normal core damage as your input for your probabilistic risk assessment. It's not -- this portion of it isn't actually trying to evaluate the FFRD.

So, okay, so I think I've covered everything on this slide, so like I said, so we are in region three for both your change in core damage frequency and the change in large early release frequencies for this, for the very small changes. All right, so next slide, please?

So, for Section 2.2.3.2, the acceptance guidelines for the self-approved risk-informed evaluations, this discusses the risk metric that could be made by an entity without NRC review and approval. So, whereas paragraph (h)(2)(ii) required you to be in the very small region, this is now, for self-approved, is now what we call minimal changes.

So, paragraph (h)(1)(ii) requires that the NRC-approved risk-informed evaluation process to demonstrate that any increase in the estimate risk is considered to be minimal. It's still building on using those Figures 4 and 5 in Reg Guide 1.174, and it does, this section does have a, sorry, does inform entities that minimal impacts on risks should not be risk-significant. That's just a basic assumption that we're starting with, and that any minimal criteria are less than two orders of magnitude below the maximum allowed increase in Reg Guide 1.174, right.

So, we're talking one times ten to the negative seventh for a change in core damage frequency as opposed to the one times ten to the negative fifth that's currently allowed for a change in region two, and it's one order of magnitude less than the very small criteria, right.

Therefore, the calculated increases must be less than your ten to the minus seventh per reactor year for core damage frequency. So, we're showing on this picture, for minimal changes, you're no longer in region two or region three. You're now below and less than this mark on the figure, so you're less than that. So, I'll stop here before I give the last part, so go ahead, please.

DR. BLEY: This is Dennis Bley. Have you had any comments from stakeholders about this?

MS. BUCHOLTZ: The comments that were back on the -- so this has not been presented to the public as of yet, other than this is the first meeting that is even open to see, so we haven't had any feedback yet from industry on this.

However, the feedback that we did get in 2007 and '08 time frame, the feedback we got was the very small was a tight band for the other band, but minimal wasn't defined at that time, so we didn't receive any comments on minimal then, so we're looking forward to seeing them, you know, in the future when this goes out.

DR. BLEY: Okay.

MS. BUCHOLTZ: Yeah.

MEMBER BIER: Hi, this Vicki Bier again.

I'm curious about what auditing or accountability is available for so-called self-approved changes. Again, I understand the reasons for wanting to have self-approved. It makes a lot of sense. If the change is small, you don't want to go through a huge amount of, you know, NRC paperwork, and review, and whatever.

But I know in the aviation industry, there have been some problems of like airlines being approved to do their own checkrides for pilots, and then found out that some places were just pencil whipping the checkrides, and so, you know, what are the consequences or accountability measures on those?

MS. BUCHOLTZ: Right, so right now, any minimal changes that are done by an entity, all of those have to be reported to the NRC every two years. So, they would, every two years, they would have to look at them together to make sure that they don't add up to a, you know, that all the minimal changes did not add up to a very small change, and then they would have to be submitted to us every two years. So, we would get that information every two years of all of the changes that they've made, and then we would look at that to see if they are using this program correctly.

MEMBER BIER: Okay, so if anything looks

really suspicious or questionable, you would be able to dig in and figure out why they had made that change?

MS. BUCHOLTZ: Yes, exactly, and it will be upon the NRC staff when we would approve the risk-informed evaluation process to ensure that the process that they've laid out, you know, it is appropriate to screen the risk assessments that they're doing for these metrics too.

So, we would ensure when we do that, just like we do for the risk-informed completion time programs, that the program is set up to, you know, to take appropriate actions based on the thresholds that they calculate, yeah, all right.

MEMBER PETTI: Kristy? This is Dave. Just to understand, you'll need to approve their risk evaluation process --

MS. BUCHOLTZ: Correct.

MEMBER PETTI: -- but they wouldn't have to, with every license amendment, with every fuel reload, have to regurgitate all of that information over and over. It's kind of just the opening volley, if you will.

MS. BUCHOLTZ: Correct, yes. Yeah, once the risk-informed, once they've submitted the process to the NRC and we've reviewed it, looked at it, and made sure that -- you know, just like we would with like the other risk-informed rules that we do where we allow them to have a process, like with 50.69 or the risk-informed completion time.

We're envisioning that they would take those programs that they've already had and build on them to use this. So, we hope that this wouldn't be, you know -- we know it will take some effort, but we're hoping it's less than, you know, starting from scratch, so if they have those programs, it would be able to build on those.

And no, and so we don't expect them to do this -- once that process is laid out, we expect them to be able to use it as they do now for the other ones, yeah, all right?

MEMBER PETTI: All right, thank you.

MS. BUCHOLTZ: Yeah, no problem. So, for the version of Draft Guide 1426 that we did submit to the ACRS, we had in there that there was a one-percent -- we had an increase in CDF and LERF of less than one percent for the overall plant risk we had in there as a limitation, and then we did decide, looking back, that we did remove that.

That was my mistake. I missed the fact

that the Commission had told us that we're not -- I was thinking of new reactors that are on the lower line, and so I put that percentage in there thinking that we would want to make sure that we're not, you know, allowing for a reactor that has such a low PRA that you could double it or triple it and still be, you know, and it not be considered significant.

But I was reminded that the Commission had weighed in on that, and that is a benefit of having a lower-risk plant in that the bar is set where it is and that would still be safe, and so I did remove that. I missed that, so you'll see that that has come out of the draft guide. Okay, so next slide, please?

So, for Element 3, Element 3 is defining the implementation and monitoring program. So, this, again, this is another section that points to Reg Guide 1.174 for the implementation and monitoring. The original rule, like Joey was talking about earlier, it did have a period updating of two things.

It had a period updating of the risk assessments themselves, where the entity would have had every five years, look at all of the changes that were made under the program and then redo the risk assessment to verify it didn't go above the very small changes when they're all put in place. We removed that from this because that's not typically done for other license amendments that we currently do.

So, every change they do, they would evaluate the risk in place at the time and the current changes that have been put into place, but we would not make them redo every five years to do that, so it would be done every time they make a change instead, right.

And then the -- so we did remove that from the rule itself and then we removed it from the draft guide, so it no longer includes that reevaluation of risk assessments.

And then we did also remove the requirement to update the PRA every five years from the draft rule. We decided that that, you know, that's an ASME standard, and that we rethought that and thought that that didn't need to be in the actual rule as a requirement, that it is in the ASME standard that they're following.

So, we did leave it -- so we took it out of the rule, so it doesn't say that you have to do it every five years, which had the intent of every two refueling cycles, but we did leave it in the draft guide.

So, in the draft guide, it's still in

there that an entity should update, you know, do any updates and upgrades to their PRA at a maximum or at the long time of every five years, but it was changed from a must to a should, right, because we expect them to still follow the PRA standard for that.

And then, of course, like the note on the bottom of the slide, it's just reminding everybody that the version of Draft Guide 1426 was still under concurrence when we submitted it, so we have made these four -- so these four changes were listed in that draft guide, but they have now been changed or removed. So, next slide, please?

MEMBER ROBERTS: Yeah, Kristy, it's not on this slide, but this section of the draft guide talks about if you have a design that doesn't meet single failure criteria, because you're allowed to above the transition break size, what the allowable outage time would be, and one of the paragraphs says it should be a pretty short time, without defining short, and the paragraph after that seems to say you can't do it at all, and so I was wondering what the thought process was in terms of allowable outage where there's basically no redundancy as permitted by the rule.

MS. BUCHOLTZ: Yeah, so the way the -- so that is -- thank you. I thought I added that here.

So, that was one of the things that we did remove from the rule.

The rule had in there, before we decided to make sure that all of the systems, structures, and components for this were placed into tech specs, we did have in there a short time, so that if an entity was operating and then recognized that they were in a condition that wasn't analyzed by the rule, it gave them a time period to basically come back into compliance, because you would be out of compliance with the rule.

So, since then, we have decided to remove the short time from the rule and we put in the requirement instead to make sure that the systems, structures, and components that are important are in the tech specs, so this way if you had a problem with a piece of equipment, it would give you the time in the tech specs to restore or to do maintenance, right.

We also recognized that maintenance, if you had to do maintenance on any of the equipment and you took it out, right, you could possibly not meet -- you know, especially if you have a realistic analysis that's assuming you have everything or it's assuming those changes, right, you might not meet that. So, we did take short time out of the rule, and so that's been deleted also from the draft guide.

MEMBER ROBERTS: Okay, so both of those paragraphs I mentioned are being deleted from the draft guide?

MS. BUCHOLTZ: They are.

MEMBER ROBERTS: Okay, thank you.

MS. BUCHOLTZ: Yeah, my apologies. I thought I added that to this and didn't save it. Okay, so Element 4. So, Element 4 is submitting the license amendment request.

So, for all of the requests under this, excuse me, for all of the requests under this rule, it's expected and we lay out in the draft guide that a summary of the PRA model and methods used to evaluate the proposed change should be in the submittal, right, and with the basics of it, you know, which risk methods are being used, why are they acceptable, right, your key modeling assumptions, your considerations of uncertainty, the key operator actions, your changes required for your event, your fault trees. That's normal and typical of any license amendment that's done under Reg Guide 1.174, so that stays there.

And then we added in there for those that choose to do the risk-informed evaluation process, that any submittal or your initial submittal for this, right, then you should have submittal details to support the changes without the approval, right.

So, that should have a description of the entity's PRA models, any non-PRA risk assessment methods to be used, right, description of the entity's approach, methods, your decision making to evaluate, right, your risk criteria, your defense in depth, your safety margins, your performance measures.

So, we do have -- so, Element 4 is divided into two sections, which it kind of has the section, or the discussion that's normally in Reg Guide 1.174, and then we added a couple extra stuff for the risk-informed evaluation process. All right, next slide, please?

MEMBER PETTI: Kristy, before you go there, this may be broader than you, the discussion about safety margins and defense in depth, going to best estimate for a large break LOCA and making it beyond design basis will open up some margin, and an entity may decide they want to use that margin in a power uprate, for instance. So, and similarly, how defense in depth looks after you execute that change is going to be different than what it was in the plant before.

So, does this all recognize that defense in depth could still be adequate? It's just going to be different than what it was before, and that yes, you're going to be giving up some safety margin, but the plant will still be safe? I just worry that people could get misled, particularly, you know, the public as they would be looking at these.

So, these are some fairly esoteric topics we've been discussing, but at the higher level, the answer is yes. It's just different in form is how I would describe it to the non-technical person. Is that a fair kind of way to think about it?

MS. BUCHOLTZ: It is a fair way to think about it, and I did bring in for this reg guide or the draft guide, I did include to say in there, there is a couple of paragraphs in there, well, I should say a little bit of discussion in there that points out that yes, that it is not the intent that you can't make any changes to defense in depth or safety margins, right. That is not the intent here.

The intent here is to look at them and to make sure that you still have appropriate measures in place for both of those, but we do recognize that yes, there might be, depending on what proposed change you're making, that the defense will, you know, the defense in depth will look different, and the same thing with the safety margins.

So, we do have that. I hope that's clear in there, and I figure public comment will let us know for sure, but we did -- we do have -- that is the intent of those sections, to say that it's not going to look the same, but it just needs to make sure you still have, you know, you're still in line with the defense in depth, you know, the philosophy of the Commission all together.

So, it might shift, you know, barriers, or it might shift systems as your defense, or you might be relying on more non-safety stuff now, right, as another layer in there, so, yeah, the defense can change, yeah. All right, so next slide, please?

So, the last part of the draft guide is the quality assurance and the documentation piece, and these two were pretty much the same. So, the quality assurance is the same as Reg Guide 1.174 with no substantial changes, again, just the changes that we talked before, changing it from licensing basis to proposed change and changing it to recognize an entity versus a licensee.

And then for the documentation piece, we did bring in -- so it does differ a little from Reg

Guide 1.174 in that we're pointing more to the documentation that's used for Section C.4 of Reg Guide 1.200, which is evaluating your changes from a base PRA perspective.

So, you would document the -- it still involves the same, much of the same documentation as Reg Guide 1.174 contains, but it goes a little further in-depth on documenting changes to your base PRA in your models that you're doing for this program, so, to ensure that everything is documented and nothing is missed, but it is consistent, like I said, with Reg Guide 1.200 and Reg Guide 1.174. There was no real intent to do anything different than what's already been established under those reg guides.

So, all right, so the next slide, please? So, that is the end of the draft reg guide. I'd be happy to answer any other questions that I didn't discuss or didn't answer on the risk portions. Okay, hearing none, that's all I have.

CHAIR BALLINGER: Okay, we are at the end of the presentations. I think we need to now go out for public comment. So, if there are any members of the public listening in or out there that would like to make a comment, please state your name and organization and make your comment.

MR. BURKHART: Yeah, this is Larry Burkhart from the ACRS staff. If you are on the MS Teams link, we kindly ask you to raise your hand and we can get to each comment in sequence.

CHAIR BALLINGER: Well, I'm not hearing anything. From the audience members or participants in the audience, would anybody like to make a comment? We probably beat everybody to death.

(Laughter.)

CHAIR BALLINGER: Hearing none, other comments from the members before we adjourn for the day? None? Well, I'm sure I speak for the rest of the members that you guys have really done a yeoman's job of putting this presentation together and it's very much appreciated, and we'll continue to inflict you tomorrow on this. And so, if there aren't any other last-minute comments, we are adjourned until tomorrow morning at 8:30.

(Whereupon, the above-entitled matter went off the record at 4:59 p.m.)

C E R T I F I C A T E

This is to certify that the foregoing transcript

In the matter of: ACRS Fuels, Materials and Structures Subcommittee

Before: NRC

Date: 12-17-24

Place: teleconference

was duly recorded and accurately transcribed under my direction; further, that said transcript is a true and accurate complete record of the proceedings.

Court Reporter

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NUCLEAR REGULATORY COMMISSION

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Docket Number:	(n/a)
Location:	teleconference
Date:	Wednesday, December 18, 2024

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

REGULATORY RULEMAKING, POLICIES AND PRACTICES

SUBCOMMITTEE

+ + + + +

WEDNESDAY

DECEMBER 18, 2024

+ + + + +

The Subcommittee met via Videoconference, at 8:30 a.m. EST, Ronald G. Ballinger, Chairman, presiding.

SUBCOMMITTEE MEMBERS:

RONALD G. BALLINGER, Chairman
VICKI M. BIER, Member
VESNA B. DIMITRIJEVIC, Member
GREGORY H. HALNON, Member
CRAIG D. HARRINGTON, Member
WALTER L. KIRCHNER, Member
ROBERT P. MARTIN, Member
SCOTT P. PALMTAG, Member
DAVID A. PETTI, Member

THOMAS E. ROBERTS, Member
MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:
DENNIS BLEY

DESIGNATED FEDERAL OFFICIALS:
ZENA ABDULLAHI
WEIDONG WANG

ALSO PRESENT:
LARRY BURKHART, ACRS/TSB
THERESA CLARK, NRR/DSS
JAMES CORSON, RES/DSA/FSCB
ELIJAH DICKSON, NRR/DRA/ARCB
DAVE GARMON, NRR/DRA/ARCB
JOHN LEHNING, NRR/DSS
JOSEPH MESSINA, NRR/DSS/SFNB

C-O-N-T-E-N-T-S

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

8:30 a.m.

CHAIR BALLINGER: The meeting will now come to order. This is a renewed meeting of the Fuels, Materials, and Structures Subcommittee of the Advisory Committee on Reactor Safeguards. We had a meeting yesterday. This is a continuation of that meeting.

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

We have consultant Dennis Bley who I think he's -- yes, he's there. Zena Abdullahi and Weidong Wang of the ACRS staff are the designated federal officers for this meeting. No member conflicts of interest were identified for today's meeting. We have a quorum.

During today's meeting, the subcommittee will receive continuing presentations on technical topics regarding increased enrichment conventional and accident tolerant fuel designs for light water reactors rulemaking. The Nuclear Regulatory Commission is proposing to amend these regulations related to the use of conventional and accident tolerant light water reactor fuel designs. The NRC's goal is to establish effective and efficient licensing of the use of fuels enriched greater than 5 weight percent uranium-235 while continuing to provide reasonable assurance of adequate protection and public health and safety.

The new requirements also would address fuel fragmentation, relocation, and dispersal in relation to the key accident tolerant fuel components of increased enrichment and burnup limits. The ACRS was established by statute and is governed by the Federal Advisory Committee Act or FACA. The NRC implements FACA in accordance with its regulations.

Per these regulations and the committee's bylaws, the ACRS speaks only through its published letter reports. All member comments should be regarded as only the individual opinion of that member, not a committee position. All relevant information related to ACRS activity such as letters, rules for meeting participation and transcripts are located on the NRC public website and can be easily found by typing About Us ACRS in the search field on the NRC's home page.

The ACRS consistent with the Agency's rule of public transparency and regulation of nuclear facilities provides opportunity to public input and comment during our proceedings and receive no written statements or requests to make an oral statement from the public. We have also set aside time at the end of this meeting for public comments. The ACRS will gather information, analyze relevant issues and facts, and formulate proposed conclusions and recommendations as appropriate for deliberation by the full committee.

A transcript of the meeting is being kept and will be posted on our website. When addressing the subcommittee, the participant 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 or by pressing *6 if you are on a phone.

Please do not use the Teams chat feature to conduct sidebar discussions related to the presentations. Rather limit use of the meeting chat function to report IT problems. For everyone in the room, please put all electronic devices on 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 and they're very sensitive. For presenters, your table microphones are unidirectional -- very unidirectional. And you'll need to speak into the front of the microphones to be heard.

Finally, if you have any feedback for the ACRS about today's meeting, we encourage you to fill out the public meeting feedback form for the NRC's website. So we'll now proceed with the meeting. And do we want to have any additional -- any intro from Theresa? Okay.

MS. CLARK: So thank you very much and good morning, everyone. I'll keep it brief. I don't have a whole speech this morning. But we really appreciated the compliments from the committee yesterday and especially the interest and engagement with lots of questions.

Though it may not always feel like the hot seat, we really do appreciate that and it shows the interest in our work. I think some of the questions go to sort of the healthy tension of any optimization project where we're going between whether it's schedule and the ability to integrate feedback, whether requirements should be more performance-based or more prescriptive, how we relax certain requirements while keep guardrails on it to ensure safety margins of defense in depth. We expected some of this conversation, and we look forward to continuing it.

We've heard a really robust technical basis for the work the staff has done. We'll hear even more of that today and the details. And we'll still have January to further link everyone to their details. So I look forward to that and appreciate the time.

CHAIR BALLINGER: All set? Okay. Whoa, we only have one person.

MR. CORSON: Yeah, you're going to be hearing a lot from me this morning. So my name is James Corson. I'm a senior reactor systems engineer in the Fuel and Source Term Code Development Branch in the Office of Research.

So I'm going to be talking about some guidance related to fuel performance this morning. The first presentation is related to some old work that dates back to the 50.46c rulemaking. So there's three draft guides on zirconium-alloy cladding analytical limits that I'll talk about first.

Later on this morning, I have a separate presentation that gets into fuel dispersal. So that will be a new draft guide. Next slide, please. So as I said, as part of this 50.46c rulemaking dating back about a decade now, we developed three draft regulatory guides relative to zirconium-alloy cladding analytical limits.

So the first two are essentially test procedures. There's on related to measuring breakaway oxidation. And I'll talk a little bit about what that is later on.

A second one relating to how you measure post-quench ductility related to oxidation and embrittlement of zirconium-alloy cladding. And then the third draft guide has to do with how you apply those test procedures to create analytical limits for zirconium-alloy cladding. So we updated these documents to reflect the changes that we made to the rule language in 50.46a.

The parts that we took from 50.46c, we made some changes to the rule language. So the guidance has been updated accordingly. Otherwise, the guides are mostly unchanged for what was submitted in the draft final rule package.

Of course, there was some minor editorial changes here and there. But for the most part, it's the same as what we submitted in 2016 with some changes to address differences in the rule language. So next slide, please. So this slide is meant to show how these guides relate to the rule language that we have now.

So Joe Messina talked a lot about the rule language itself yesterday. So we had these fuel system requirements that we need. So the first one, of course, is to address cladding degradation phenomena.

That's essentially what these three guides are getting at. There is this hydrogen limit in one of the guides that we put there because we removed the limit from the language itself. But otherwise, these guides are mostly meant to address cladding degradation phenomena.

However, the point of addressing these cladding degradation phenomena is to prevent widespread brittle failure of the fuel and cladding. That could challenge fuel coolability. So ultimately, these are meant to ensure fuel coolability.

Now as you'll hear later on this morning, just addressing cladding embrittlement is not sufficient to address fuel coolability because of fuel dispersal. But these guides do try to address this issue. So next slide, please. On this slide, I wanted to highlight some of the zirconium-alloy cladding degradation mechanisms that there are.

So I'll focus on the two lower things in this figure. So loss of ductility, this is just due to oxidation, oxygen uptake that leads to cladding embrittlement. As you see on the top right, this is a figure showing how this embrittlement changes with the hydrogen content of the cladding.

So that's one mechanism. Another mechanism that occurs more in small break type LOCA accidents is breakaway oxidation where prolonged exposure at lower temperatures can lead to this behavior which I'll talk about a little bit more. But one of the other things I want to highlight on this slide is that there are other cladding degradation mechanisms out there.

Our guidance does not address them because these other degradation mechanisms, the loss of ductility due to brittle failure and breakaway oxidation happen at lower temperatures. So setting certain limits to address these also addresses these higher temperature phenomena. But this is just to show that if you ever wanted to go to higher temperatures, there are other things that would need to be addressed.

CHAIR BALLINGER: This is Ron Ballinger. I just feel compelled every time I see this, I make the same comment. That figure on the lower right is not from any U.S. cladding.

It's from E110. It's Russian cladding. So it's almost deceptive to put that there as an example of breakaway oxidation. You would think that you'd be able to find a photo where they have real breakaway oxidation on Zircaloy-4 or ZIRLO or something like that. But I suspect it doesn't exist.

MR. CORSON: So there are photos not as extreme as this but where you start to get discoloration and you get significant hydrogen uptake which is what the real concern with breakaway oxidation cladding is you're right, yes. This is E110. I think one of the important things that I'll get at very shortly is that there are a lot of manufacturing processes and minor changes to the alloy that can have a significant impact on breakaway oxidation.

So this is E110. E110 has pretty much the same composition as M5 except it's made in a different way. And the way it's made is what leads to this behavior whereas M5 behaves very well when it comes to breakaway oxidation. So yes, that --

CHAIR BALLINGER: And this is a public meeting. And we really do want to be careful that we don't send a message which we don't want to send.

MR. CORSON: Yeah. No, you're right. U.S. alloys as far as we've seen so far behave very well when it comes to breakaway oxidation. But the trick is you can make minor changes to how you make that cladding that can turn it into something that looks more like this. That hasn't been done in the U.S. But it could theoretically happen. Next slide, please.

MEMBER PETTI: So just a question here. As I first looked at this and particularly the plot on the upper right, just you know how much Zircaloy has been tested in the old days. And the old days aren't that old for some of us.

And yet there's all these new mechanisms. That's how it's described. I interpreted that to mean they were new because we ended up going to higher burnup, higher hydrogen content. But you seem to be saying it's an issue because we didn't study oxidation at the lower temperatures. Explain to me sort of how we got here.

MR. CORSON: So the breakaway oxidation thing is something separate. So again, this happens at lower temperatures, prolonged exposure. It was first discovered because someone was doing testing on E110 cladding and saw this happen.

And so that caused the NRC and other organizations to study this further. And as I had said, we didn't see the same behavior, certainly not at the short times that E110 experienced breakaway oxidation. So the top right, that's more the traditional higher temperature oxidizing at 1,200 C per 2,200 degrees Fahrenheit and studying the embrittlement. And so that -- yes, that is related to higher burnup phenomena and hydrogen uptake in the cladding.

CHAIR BALLINGER: Excuse me. Somebody has got a mic open or something. Got it? Thank you.

MEMBER PETTI: Okay. So this is the other problem I'm having. There are no Reg Guides telling vendors how to manufacture their clad or tests they need to do to prove that the clad is okay when we don't have any indication that any U.S. clad experiences this behavior. So that you would -- and why wouldn't we just somehow deal with it as a specification?

Again, this just jumps out at you. And again, this was all done back, I guess, around 2000s.

How did we get heat to that position?

MR. CORSON: So yeah. So basically, there are manufacturing changes that can lead to this type of behavior, that can turn a good performing alloy into a bad performing alloy. And it's even some minor things as potentially where you're getting your Zircon from. Like, it could be down to the trace impurity level.

And so at what point does the NRC or does the vendor want to restrict their process to using a certain time of material, source it from a certain place, use a certain process to refine the Zirc, use certain surface finishing techniques. So there's a lot of steps in the process that could potentially be controlled. Or you could come up with a relatively simple test just to show that nothing bad has happened, that you haven't made some minor change that turns into a bad performing alloy.

MEMBER PETTI: Although it seems like a relatively simple test, fuel vendors don't have furnaces to do oxidation testing in their back rooms usually.

MR. CORSON: Yes, so --

MEMBER PETTI: I mean, they're used to saying, okay, we're going to inspect the impurities to a certain limit. We're going to inspect processes or put some process specs on it. I mean, that's -- my sense of fuels, that's sort of where they're comfortable. This just seemed a little bit out of the ordinary. But if industry thinks this is okay, I guess I'm okay with it. Just doesn't --

CHAIR BALLINGER: Following Member Petti's comment, we wrote a letter on this on 50.46c which has been -- the whole thing has been withdrawn. And we made the same comments. And in the original documents, there was literally a requirement for vendors to actually do the testing on every batch, if I recall. And looking at the new rule that's not there which is a good thing, I think. So that's one, in my opinion, one person's opinion, good changes to the new rule.

MR. CORSON: So what I'll say -- I mean, I'll get into this a little bit later. But I might as well say it now is that part of the guide is saying the vendor should come up with a plan for how often they're going to test and how they're going to show that they haven't made changes that adversely impact breakaway oxidation behavior. And so that plan would be reviewed and approved by the NRC.

And so they could very well propose

infrequent testing. Every once in a while, we're just doing a sanity check, so to speak. So that could be done. It would be reviewed and approved by the NRC.

There is a default time frame in the guide which is once per ingot which is actually fairly often. But again, that's just a default. The vendors can come up with their own plans based on the types of arguments that you're saying right now are based on, okay, this is how we're controlling our process. So we're fairly confident that we're not going to change anything that messes it up. So that's part of the back and forth with vendors.

CHAIR BALLINGER: Because we visited one of the vendors. And we asked them about the cost, and it was going to be passed on to the customer, each load. And it was about half a million dollars. So it's not chump change to do these kind of tests to verify.

MR. CORSON: Yeah, I mean, it's tricky because we don't understand all of the mechanisms and all of the changes that can lead to this behavior. So this is why we had this. There was a testing requirement in the actual rule language for 50.46c.

That testing requirement is no longer in the rule language itself. So that does provide some amount of flexibility right there. You can always argue that it's not necessary for the reasons that you're saying.

MEMBER PETTI: So let me ask a different question. Has anyone tried to manufacture optimized ZIRLO or M5 in a bad way to actually get the breakaway oxidation?

MR. CORSON: Not to my knowledge. So I know -- so at Argonne National Lab during this process, they focused a lot on the different surface finishing mechanisms. They looked at different types of acid baths or polishing and so on and identified some of the things that could lead to worse behavior.

Argonne did not do this. But I believe it was TVEL, the Russian manufacturer or some other organization looked at E110 and sourcing it from western styles, Zirconium sponge. So using the Kroll process and everything else. And that made it perform much better.

So one of the things that was speculated is that the impurities that were removed during their electro-refining process that they had used to make E110, some of those things that removed actually helped breakaway oxidation, provide more favorable breakaway oxidation behavior in western alloys. So yeah, no one tried to make optimized ZIRLO to make it bad, but someone did try to make E110 to make it good. And that's essentially what they do now for E110. They know enough to make it perform better.

CHAIR BALLINGER: Dave, I think what we're getting at here is that this is a case where we're running the risk of the tail wagging the dog. You know what I mean. We see bad behavior, terrible behavior actually, and then infer that terrible behavior is somehow a done deal. It's going to happen with our material.

And yet we haven't been able to make it do that. And yet we've sent our material, our sponge, over to Russia. And lo and behold, they can't make it happen. So it's just a worry that there's a burden here that is pretty costly.

MR. CORSON: Yeah, I mean, it's something, I think, we'll have to think about moving forward when we get into the draft final rule stage. I think to be honest, part of the reason why we're here is that we try to make as few changes as possible from 50.46c. So we just -- we try to focus on the things that really did change, the new stuff like fuel dispersal and everything else.

It was sent to the Commission. We dusted it off, made some few changes, and sent it back out. So yes, obviously, I think this is something that we will need to think a little bit more about and perhaps make it better.

MEMBER PETTI: Yeah, I mean, I would love to it. The Reg Guide says in your process, you cannot use fluoride-containing material, stuff that you know causes the problem. You just say, you can't do that.

I think a manufacturer can deal with that easier than this. If you can figure out what the real issue was which means you got to dig a little deeper. I mean, so I don't know. I conclude did you guys decide that it wasn't worth doing more of the research to figure this out or that there was some statistical thing and you're chasing your tail? You could never tie it down in terms of what's going on.

MR. CORSON: Yeah, I mean, I was getting at this with different sponges. So there's zirconium sponges. So there's different trace impurities. We don't know exactly which of those trace impurities are the ones that are problematic or good.

So trying to figure all that stuff out, we didn't keep drilling down. And also then there's -- I guess the question is -- and this is maybe something for industry later during the public comment period.

Is it better to tie down your process and to restrict it in the ways that you're suggesting?

Only do these certain things. Or is it better to do a test. And so that -- it might be useful to have some feedback during the public comment period on this issue. Do you want restrictions on your process or do you want a test and not have to worry about all the things that are happening?

MEMBER PETTI: This is the 21st century. And I think manufacturers make stuff a heck of a lot better than 20, 30 years ago, so --

CHAIR BALLINGER: I think that during the discussion on 50.46c, that question actually came up. And I think the consensus was on part of the industry was that they would control the manufacturing process which is what they already do. But the consensus on the part of the staff, we want you to do a test.

And again, not to beat a dead horse, the vendors don't want to see this. If this were to happen once on one load of fuel, anytime they would be in deep yogurt, so to speak. So there's a very strong impetus to make sure that they control their process so that something like this does not happen. Just one person's opinion.

MR. CORSON: No, I mean, I went back and

read the transcripts for the 50.46c ACRS meeting. So I was aware of this back and forth. I think it is difficult without knowing all the things that could make this happened to restrict your process in such a way that it wouldn't happen.

But I think it is a good point. Like, do you need to do any testing at all, which I think that it's a little risky without knowing all the things that could happen. Do you do a test like very, very infrequently just as a sanity check? Or do you do it more often? I think the way the guide is written now is probably leaning towards more often with some flexibility to have more infrequent testing. But yeah, that's something that we can think about and revisit.

CHAIR BALLINGER: Now we're probably an hour behind.

(Laughter.)

MR. CORSON: Well, the good thing is, I mean, we felt good.

(Simultaneous speaking.)

CHAIR BALLINGER: We feel good about it.

MR. CORSON: Yeah, the good thing is I think I can probably skip through some of the slides on that guide, but we'll see. So next slide. Okay.

So this is an abbreviated history of just the whole 50.46c.

I think what I wanted to show here is that there was a lot of back and forth. We're aware there were a lot of industry comments on the original versions of these guides, some of which we reflected in what was submitted in the draft final rule package for 50.46c. So we did get some industry feedback along the way that led to changes from what was in the original version of the draft guides that were sent out in 2014. So next slide, please.

All right. So I got into some of this. But basically, breakaway oxidation is you have normally under LOCA conditions this protective oxide that forms -- that protective oxide, the tetragonal oxide is metastable. So there's things that can cause it to switch to this nonprotective monoclinic oxide which leads to higher oxidation, higher hydrogen uptake.

And so that's the bad stuff that we're talking about later. And as I said, we showed that some minor changes in alloy composition or manufacturing process can have a pretty significant impact on the breakaway oxidation behavior. So the E110 that we were talking about a little -- or a lot before, that showed that behavior in about 500 seconds.

So pretty quickly relatively speaking whereas western alloys, we're talking more like 4,000 or 5,000 seconds. And then the behavior is not as extreme as for E110. So that's just an order of magnitude I'd say better in terms of timing to this. So next slide.

So the guide defines an experimental technique for doing the test. It has some flexibility for doing it in different ways at different labs. And then it discusses both the initial testing and then the confirmatory testing.

So as I'll talk about later, the initial testing that was done at Argonne is sufficient for the existing alloys. The periodic testing I think is what the controversy is about, how often you would have to do the test and so on. So next slide.

MEMBER PETTI: So chrome-coated clad, where does that fit?

MR. CORSON: So yeah, chrome-coated cladding in general is not really covered by these guides. So we do have interim staff guidance that's related to chrome-coated cladding. And it has a few things to say about how you might go about addressing some of these things, one of which is just to say, hey, this behaves about the same as existing cladding. Or it might behavior better but we're not going to take credit for it.

And so there would probably need to be either some testing or some engineering arguments, at least to address the post-quench ductility stuff. For the breakaway oxidation, it might be a little bit simpler. But yeah, so the guides don't really cover coated-cladding, but they may be informative.

CHAIR BALLINGER: This is Ron Ballinger again. But I think the trend is to go for all the BWRs, for PWRs, or all the venders that use chrome-coated cladding. Is that correct?

MR. CORSON: I can't comment on what the industry wants to do. I mean, they haven't submitted any license amendments. Certainly all the vendors are still doing testing and development plans to develop these claddings. Whether they get adopted or not, I leave that up to --

CHAIR BALLINGER: Well, they have LTAs.

MR. CORSON: Yeah, they do. I mean, they need that in order to study whether this is a viable cladding or not. And they would need for NRC approval. Whether it gets adopted or not, I mean, that's going to come down to what the industry wants to do. Whether it makes economic sense for them, safety sense, whatever else, I can't really say how that's going to go.

MEMBER KIRCHNER: Ron, this is Walt. Just an observation, wasn't this rule to include accident tolerant fuel?

CHAIR BALLINGER: That's what I was getting at, yes.

MEMBER KIRCHNER: Yeah, I'm a little concerned. Just an observation.

MR. CORSON: So the way the rule language is written, it does allow for accident tolerant fuel. They're generic criteria. The guidance does not necessarily apply to the different accident tolerant fuel concepts. And the reality is in order to do that, there would need to be some sort of testing to show that, hey, this guidance is applicable or it's not. And so in order to do that, that would take a significant level of effort which we just did not have for this rule.

CHAIR BALLINGER: But I think you do. Forget the DG numbers. But there's a progression of what you need to do if you deviate from especially radiation effects because the radiation data is for

Zircaloy-4, ZIRLO.

But if you deviate from that, the amount of work you have to do depends on the amount of deviation from that database. And chromium coated fuel would definitely be deviating from that database. So would that not -- as far as the radiation effects and the like goes, wouldn't that be covered?

MR. CORSON: So the challenge is that there may be -- and, in fact, there probably are different cladding degradation mechanisms. So in particular, you can get the fusion of the chromium into the zirconium that forms a brittle inner metallic. And so this diffusion, of course, it's going to be a function of temperature.

At higher temperature, you'll get more inner diffusion. How does that impact cladding embrittlement? Certainly, I think you can do the test procedure.

You can apply the test procedure that I'll talk about to show whether you get the same calculated Cathcart-Pawel equivalent cladding reacted and how the embrittlement works as a function of this curve. But again, you have a different mechanism. And so it's hard to say whether the existing testing would necessarily be sufficient to cover all these different degradation mechanisms.

CHAIR BALLINGER: Yeah, I think the existing testing would not. But the guide would require you to do the additional testings.

MR. CORSON: Yes, yeah. So I think that the guidance, in terms of the test procedures, could probably be used to provide some arguments for why this is safer. It meets certain requirements. But you would also have to address these different degradation mechanisms.

And that's essentially what the interim staff guidance on chrome-coated cladding says. It basically says the existing stuff may be informative. But there are other things to consider, another one, of course, being the chrome zirconium eutectic which takes place at a higher temperature than what we're talking about here. But it's still not so much higher. It is something to consider.

CHAIR BALLINGER: We never said it was going to be easy.

MR. CORSON: Yeah, definitely. Definitely not. Next slide, please. Okay. So I was talking about this before. The vendors would come up with their own plan for how often they would do the testing.

The guidance allows for relaxation of test frequency with time. And then they wouldn't submit the test results. But they would be available for audit.

So we talked about all this before except for that. That last point, I didn't mention. So next slide. All right. So that's it on breakaway oxidation until we get to 1263 about how you apply that limit.

So 126w is determining post-quench ductility. And so this is coming up with -- or this defines an experimental method to measure the ductile-to-brittle transition for the zirconium-alloy cladding. So historically going back to the original ECCS rulemaking, we try to define the success criteria in terms of maintaining some cladding ductility.

So that's what this testing is for. So the technique has some flexibility where possible to allow for some different variations in equipment. Both this procedure as well as the one in 1261, they started out as Argonne National Lab test procedures and then they were expanded a little bit based on industry comments on the original 50.46c rule. So there is a little bit of flexibility there.

CHAIR BALLINGER: So these ring tests have been done for many, many, many years.

MR. CORSON: Yes.

CHAIR BALLINGER: So this is not really that controversial.

MR. CORSON: Yeah, yeah. It'd not that controversial. So I'd say this part of the guide, this part of the 50.46c rule, not very controversial. So yeah, the whole point is saying, like, at a particular hydrogen level, this is how much oxygen uptake in your cladding leads to brittle behavior.

And so the important thing is that in order to use these techniques and these guides is that you use the calculated equivalent cladding reactant using this Cathcart-Pawel correlation. So this is -- it's just a surrogate measure for the embrittlement mechanisms that are going on in the cladding. But it's a useful surrogate.

It compares really well to embrittlement. So next slide. So this slide and the next one just show how you would take your test results and come up with what that ductile-to-brittle transition is for a given hydrogen level. So one of the ambiguities in the original guide that was sent out in 2014 before the public comment period is that it's really hard to charge zirconium-alloy cladding to a given hydrogen level.

There's always going to be some variability. So there are different ways that we can come up with -- in that data together to come up with a limit. So the first approach was to take one sample.

So the way the guide recommends is to take a sample. Make it big enough that you can cut it into three smaller samples for the ring compression tests, and then measure what the offset strain or the permanent strain is. So the first approach for coming up with your limits is to just bin all three of the rings from a given sample, average them together, and that's what your average offset strain is for that average hydrogen content.

And then you come up with a brittle set or a brittle bin, a ductile bin. And then you can interpolate between the two to get what your limit is at the limiting CP-ECR. So that's one approach. Next slide.

And the other approach is just instead of binning, you just take each information ring compression test separately and fit a line through it. So those are the two approaches. And so this would be done for each hydrogen level or a certain number of hydrogen levels to come up with a curve. So next slide, please.

So how do you apply these limits to your zirconium-alloy cladding? That's what this guidance addresses. So it describes an approach to establish limits for both post-quench ductility and for breakaway oxidation.

On top of that since we move the peak cladding temperature limit from the rule language to guidance, we made it clear in the guidance that all of this is predicated on having a peak cladding temperature of 2,200 degrees Fahrenheit. If you would go beyond that, you would need to address different mechanisms. There's no guidance available now for how to do that. It's just kind of a warning. You go outside of this, you would need to come up with some other justifications and so on.

CHAIR BALLINGER: And there's also a lower temperature limit, right?

MR. CORSON: Yeah, so --

CHAIR BALLINGER: Maybe ten -- I forget the exact number. But there's a knee in the curve.

MR. CORSON: Yeah, you'll see that in a little bit. So I'll talk about that when I get to it. So the guidance also has some ways to consider the impact of oxygen diffusion from the inside surface of the cladding.

This also dated back to the LOCA research program and the 50.46c days when you have high burnup fuel and binding between the UO2 and zirconium. Some of that oxygen from the UO2 can diffuse into the zirconium which also leads to embrittlement. So the guide talks about how to handle that.

Of course, these limits for post-quench ductility are based on hydrogen content in the cladding. So you need to know what your hydrogen content is for a given burnup level or a given oxide thickness. And so the guide has some default cladding hydrogen uptake models that industry could apply if they don't feel like coming up with their own model. And then lastly, it has the analytical for combustible gas generation. That's just taken directly from the old 50.46 rule language and placed into guidance with no additional this is how to come up with a different limit.

CHAIR BALLINGER: Okay. None of your slides show the data on hydrogen uptake from which the percent is achieved. The one on M5, you don't have a backup slide with that.

MR. CORSON: No, I'm sorry.

CHAIR BALLINGER: That data looks like a random walk. And somebody put a line through that data. And it could just as well been vertical because the data is clustered around a column.

And yet we designed a 15 percent hydrogen update which I think the M5 people would disagree strongly with because that puts it almost the same as other zirconium alloys from which M5 was supposed to be better. So I'm just -- I mean, I know there's limited data. But you said that vendors could come up with a different number. But I suspect that's one where they'll come up with a different number.

MR. CORSON: Yeah, so I mean, they're free to do that. One thing I will note, this is the amount of hydrogen based on how much oxide you form during normal operation. M5 performs much better in that respect. So it has a much lower oxide thickness which means even when you consider the hydrogen pickup fraction, it still typically has a lower hydrogen uptake.

CHAIR BALLINGER: I'm just saying the 15 percent takes away a lot of their margin.

MR. CORSON: Yeah, again, this is -- this was added back, again, 50.46c days. It was added to offer something for vendors and for licensees to come into compliance with 50.46c relatively quickly without having to go for approval for some of these sorts of things. So they certainly -- Framatome certainly has flexibility and can propose their own uptake model if they wish.

Because the reality is they have a lot more data than what we do. And that was the problem with coming up with the M5 hydrogen update model was there's a lot less publicly available data for that than there was for Zirc 4 or ZIRLO. So that was, I think, part of the reason for the maybe conservative line that was drawn.

CHAIR BALLINGER: When I looked at it, I would say once we give people something or tell people something, it's hard to get it back.

MR. CORSON: Yeah, fair enough. Okay. Next slide, please.

MEMBER PETTI: So this just seems like we're solving yesterday's problem. These things are all developed back then. And because this schedule, we're going to just cut, paste, put them in here now. But it doesn't recognize -- I mean, those days, did M5 even exist in the early 2000s?

CHAIR BALLINGER: I think so.

(Simultaneous speaking.)

CHAIR BALLINGER: Yeah, 2005.

MR. CORSON: Yeah, it was part of the NRC LOCA program. We did have M5 as part -- we didn't have Op ZIRLO. But we adopted it based on some comments from Westinghouse.

MEMBER PETTI: I just -- I'm always worried when you do something like that because the whole context back then is given the context today, you're probably going to get a lot of comments over the comment period. But it's just not -- it's not a good look for the regulator, I guess is my sense.

MR. CORSON: Yeah, so the challenge is these are known degradation mechanisms. And it's important to address them in some way. How you address them, of course, is it takes time to come up with the guidance.

And so the easiest thing to do given our limited time was to use what we had done in the past. I think it will be an interesting question whether this stuff matters or whether it's more the fuel dispersal stuff. That's really what limits your design and your LOCA analysis in the end.

We'll get to that a little later. All these things are happening at 1,200 C or so. Ballooning and burst happens way before that. So what ends up being limiting, it will be hard to say.

And maybe you're right. Maybe this is going to be more yesterday's problem. And it will be other things that are more limiting that we need to be more concerned about.

So this is the acceptable limit for the currently deployed alloys. As I had mentioned, we had certain alloys that we tested in our LOCA program which included Zirc 2, Zirc 4, ZIRLO, and M5. Op ZIRLO wasn't included, but Westinghouse had submitted a public comment showing that it should be included.

So all those currently approved alloys, this limit could be used. As you had mentioned, there's this knee in the curve above 400 weight ppm of hydrogen. This has to do with the fact that you're at a low enough embrittlement level that the way you do the test, you have to ramp up to a certain temperature.

The ramp rate was slow enough that you embrittled before you hit that, that 2,200 degrees Fahrenheit plateau. So that has to do with how the testing was done at Argonne and the NRC program. And that's why there's this knee in the curve there because you hit these embrittlement CP-ECRs before 2,200 degrees Fahrenheit. So next slide, please.

MEMBER PETTI: But in reality, most of the clads today to the left of that --

(Simultaneous speaking.)

MR. CORSON: Yeah, yeah. Certainly, in the earlier figure where I show the degradation mechanisms, it was kind of small. But it's in the guide. It shows, like, burnup levels for some of the cladding that was tested.

The M5 that -- I forget the burnup. I think at least 63 gigawatt days per MTU, it was at 150 weight ppm. So yeah, you're right. And Op ZIRLO performs better than ZIRLO as well when it comes to the oxidation.

CHAIR BALLINGER: Right. But that's measured hydrogen. If you do the analytics with that rule that you have for M5, you find out that it's not the same. In other words, the required reduction in ECR for M5 gets much lower on that curve if you use the 15 percent.

MR. CORSON: Again, it's based on what your calculated oxide thickness is. And then you multiply by the pickup fraction to get what the hydrogen is. I can't say what all the calculated thicknesses are for oxide at end of life for various plants that use M5. So I can't say exactly where it is on the curve. But I still suspect it's on the left part of it still.

CHAIR BALLINGER: It's definitely on the left. But it's not 17 percent. It's more like 12.

MR. CORSON: Yeah, certainly, yeah. But I mean, that's the reality that with a relatively small amount of hydrogen uptake, as you said, maybe 100, 150, 200 weight ppm. You still get significant embrittlement behavior.

That's what all our testing has shown. So it's a real mechanism. So next slide. Oh, there we go. So this section of the guide deals with how you would apply this figure or adopt it for new alloys.

So there would still need to be some testing to show that this figure would apply to a new alloy. But it's a relatively small amount of testing compared to what was done to come up with a figure to begin with. So you do some tests on as received cladding to come up with what the fresh no hydrogen content CP-ECR is.

You do it on unirradiated cladding. That's within 100 ppm of whatever maximum calculated hydrogen content. You would get an end of life. You're do it halfway, what you get at about halfway through or half of what the maximum hydrogen uptake is. And then potentially, you do testing on irradiated material. And so the next slide will cover when testing on irradiated material would be necessary for the guide.

CHAIR BALLINGER: This is the progression I alluded to when we were talking about --

MR. CORSON: Yeah.

CHAIR BALLINGER: -- chromium coated at some point.

MR. CORSON: Okay. So the question of what is similar to alloys tested in the LOCA program. So one of the things that the NRC LOCA program at Argonne did was show that pre-hydrided cladding is a good surrogate for irradiate cladding at the same hydrogen level. So what the guide says is that if you have certain specifications, our testing at Argonne gives us enough confidence that irradiated testing would be necessary because it's similar enough to assess at Argonne that unirradiated testing could be used.

And so here are the requirements, that it's manufactured essentially the same way using the Kroll process. Operates at less than or equal to the maximum fluence tested in the Argonne program which it was cladding, I think, up to 73 gigawatt days per MTU.

So I'm not -- don't know quite what the conversion is to fluence, but it's around there. That includes similar elements to the materials that were tested and that the alloying content is similar. And it says what the deviation can be to be similar. So next slide.

CHAIR BALLINGER: Well, excuse me. Let's go back. The Russian alloy, E110, is basically ozonide, basically zirc-niobium. M5 is basically zirc-niobium. When we -- 25 percent deviation in the alloying elements, that's a lot when it comes to running back to this breakaway oxidation.

MR. CORSON: Yeah, so this is just for the post-quench ductility part. So we're talking about the post-quench ductility. So breakaway oxidation, very sensitive to manufacturing, very sensitive to alloy composition.

What we've seen for oxidation, post-quench ductility not the same thing. The curves that we've seen, pretty much all the alloys kind of line up on the curve. There are some minor differences between Zirc 4 versus M5 or whatever when it comes to post-quench ductility.

But they're pretty similar for pretty different alloy compositions. So next slide. There are situations where perhaps industry would want to come up with their own curve for their own post-quench ductility limits either because a new alloy behaves better or because we've shown in our testing program that embrittlement happens more slowly at lower temperatures. So if you had an ECCS design, that gives you lower peak cladding temperatures and you wanted to take credit for that, you could do testing at lower temperatures.

And so you would do similar sorts of testing to what we did in the Argonne program where you would look at different hydrogen levels and so on. And it provides guidance for doing that. So next slide. So we talked a little bit about the hydrogen pickup models.

As I said, the guide has some acceptable hydrogen pickup models. But of course, vendors and licensees can propose their own. Next slide. So this last part for demonstrating compliance for post-quench ductility, the point is to identify the different conditions and assumptions that maximize the predicted PCT and local oxidation.

And so it's important to do this both for stuff below the transition break size as well as above the transition break size. Demonstrate that your calculated peak cladding temperature and maximum local oxidation are below these limits. It does talk about dividing up your core into looking at different burnup levels since, of course, you have different embrittlement levels for different cladding burnup levels.

So you can take credit for that, if desired. And it has some allowances to use this figure for legacy fuel that might be in the spent fuel pool because plants occasionally will look to the spent fuel pool to put higher burnup assemblies on the exterior of the core or to deal with things like leakers and so on. And of course, if you had something sitting in the spent fuel pool that was made some time ago, it's kind of hard to do testing on that material.

So yeah, it provides allowance to do that kind of thing. So next slide. So this next section deals with breakaway oxidation. As I had mentioned for the existing alloys, you can adopt the Argonne data and then just show that whatever the either -- the measured breakaway oxidation time or you can set a more conservative limit.

So let's say if you had measured breakaway oxidation of 4,000 seconds, you could say, well, we're going to set the limit at 2,000 seconds just to give us some buffer. In your analysis, you would just show that the time spent above 800 degrees Celsius is less than whatever that minimum breakaway oxidation time is. So next slide.

CHAIR BALLINGER: Again, now I'm remembering what the curve looks like for the oxidation. And what you're calling breakaway oxidation as opposed to the little curlicues on the fuel cladding, it's really a change in the slope in the oxidation rate.

MR. CORSON: Yeah, it's --

CHAIR BALLINGER: And it's not -- it doesn't go vertical. It's just a change in the slope. I'm guessing if I recall a factor of two or so in the slope. So could somebody just include that in their overall analysis?

MR. CORSON: So one of the challenges, it's not just that the oxidation rate goes up which it does.

CHAIR BALLINGER: The hydrogen.

MR. CORSON: It's the hydrogen uptake.

CHAIR BALLINGER: But still, you could include that, right?

MR. CORSON: Yeah. I mean, I guess you

could include that. It would certainly be a little more complicated to do that sort of mechanistic analysis. It's not in the guidance, but someone could certainly propose that as being acceptable.

CHAIR BALLINGER: Because I think -- coming back to above the transmission break size where you're allowed to use best estimate and all kinds of other -- there's easier path. Let's put it that way.

MR. CORSON: Yeah, the thing I'll say, though, about breakaway oxidation is it's more of a small break phenomena. You don't really get these long temperature plateaus in the larger breaks. It really only happens in the smaller break where we have the more traditional LOCA analysis. So that's, I guess the caveat here.

CHAIR BALLINGER: Okay.

MEMBER ROBERTS: Yeah, I think you said this. But if you're looking at a no burst criterion that's around 800 C or maybe a little bit less. And I'm just wondering if this would ever come into play.

MR. CORSON: Well, again, for a small break LOCA, you may be at an elevated pressure. And so then your burst temperature could be quite a bit higher than 800 C. I mean, if your system pressure stays comparable to whatever your rod internal pressure is, you're not going to balloon and burst.

So it could come into play for lower burnup rods. For higher burnup rods, yeah, it's probably not going to come into play because they have higher rod internal pressures. But again, it just depends on what the pressure response of the system is as well.

MEMBER ROBERTS: Okay, thanks.

MR. CORSON: John Lehning from the staff.

MR. LEHNING: This is John Lehning from staff. And I just wanted to add that that no burst criterion that was being discussed was particularly for the higher burnup fuel that might be susceptible to fragmentation in that burnup range. And so it wasn't that we were suggesting that that criterion would apply for the small break, for all burnups, including the fresh fuel and so forth.

MR. CORSON: Yeah, thanks for that clarification. I'll get to that more when I go into the fuel dispersal part. But yeah, that's another important thing to note.

Okay. So this is something that we added since we moved the hydrogen -- removed the hydrogen generation limit from the rule language. We put it into guidance. Again, we didn't make any changes to come up with a new criteria.

But this is -- as far as I know, this isn't really a limiting thing. It's not controversial. So we're not changing anything there. So next slide.

I talked a little bit about this inner diameter oxidation issue. So we already account for inner diameter oxidation after balloon and burst. So once you balloon and burst, oxygen can get in the cladding.

This is already accounted for in models. So the guide just makes clear that you should still keep doing that. What's new is that for higher burnup fuel where you have this fuel clad bonding layer, you can also get some additional oxygen uptake.

What we say in the guide is that to be conservative, you just do two-sided oxidation for any higher burnup rods. So that way, you don't need to know mechanistically what's the bonding between the fuel. How much fusion do you get and so on? If you just do two-sided oxidation to account for this, that's sufficient.

CHAIR BALLINGER: Now, yeah, in Appendix -- the old Appendix K, still Appendix K, you do two-sided oxidation. But that's after you burst.

MR. CORSON: Yes.

CHAIR BALLINGER: Okay. And that's legitimately you got water and oxidation and everything. But does allowing two-sided oxidation using those rules, does it penalize people? I don't know if the oxidation rate is for a bonded system. But it seems to me like it would be lower.

MR. CORSON: Yeah, it's definitely --

CHAIR BALLINGER: A bit lower.

MR. CORSON: Yeah.

MEMBER PETTI: Solid diffusion problem, that's always lower.

CHAIR BALLINGER: Yeah, it is. So is this a significant penalty requiring that?

MR. CORSON: Yeah, I mean, it's certainly very conservative, I think. Again, this was done rather than come up with the mechanistically this is what the diffusion between solid O2 is and solid zirconium for a given burnup level. So the real diffusion is less than what we're proposing.

How much less, that's what we don't know exactly. As you say, it's probably a lot less. But this is just something to get started. And people could come up with something better based on their own data and arguments.

CHAIR BALLINGER: But if you require two-sided oxidation, you can basically double the layer.

MR. CORSON: And to be clear -- so one thing I should make clear. For the oxidation from the ID, that's just for the embrittlement. It's not for, like, added heat or other things like that.

It's just to account for extra embrittlement due to oxygen uptake because that's what's embrittling the cladding. Next slide. Okay. So that's it on the --

CHAIR BALLINGER: You made it this far.

MR. CORSON: That's it on the 50.46c stuff. So we had these three guides that we had proposed back in the 50.46c days. We made some minor updates, not many, but some minor updates to address the new rule language.

And so yeah, there was a lot of back and forth with industry stakeholders during 50.46c. I think the post-quench ductility stuff was I'd say less controversial or not really controversial. As you've said, there is some continued industry interest in the breakaway oxidation stuff that we'll have to address moving forward. So that's it for this presentation.

CHAIR BALLINGER: Okay.

MEMBER PETTI: Just a final question.

Have you guys taken any of your codes and actually try to do some analysis here of following the recipe if you will in the Reg Guides, particularly in light of the transition break size? If you think it's a small break, it could be doing some different stuff. Have you guys actually tried to analytically look at that or are you thinking about it?

MR. CORSON: So I -- so I'll start and then I'll defer to my NRR colleagues. But we do periodic analyses looking at the current fleet and comparing them to these 50.46c criterial. So some of those things kind of are in these analyses. John or Joe Messina, I don't know if you want to say more about this.

MR. MESSINA: Yeah, we do annual safety assessments -- Joe Messina from the staff. We do annual safety assessments comparing to the 50.46c remnants based on the annual reports that people under LOCA EMs for the changes and errors that they report annually. We would look at the changes that would be expected in the ECT, and the effect that it would have on our ECR when we consider -- when we compare it to --

MEMBER PETTI: That's under the component rule set. But under this new transition break size rule set, have you re-looked at everything and said, okay, what's our best estimate look like? And then now the low transition breaks, have to look at those LOCAs which maybe we haven't spent as much time looking at them in the past and look at it relative to the Reg Guides.

MR. MESSINA: We haven't done any specific calculations regarding connecting it.

MR. CORSON: So what I will say, I'm actually going to talk about this a little bit later in the context of fuel dispersal. But about ten years ago when we were considered whether to include fuel dispersal as part of 50.46c, we did a number of calculations looking at how much fuel you would expect to disperse from different plants. And for those analyses, we use our thermal hydraulics code.

The types -- more of the types of assumptions that we would use for above the transition break size, not all the same. We still, I think, did loss of offsite power at the start of the accident. But we had more nominal conditions.

We weren't necessarily biasing things to be conservative or 95/95 or anything like that. And for those calculations for some of the plants we looked at, we had peak cladding temperatures that were pretty low, like, well below where you would even need to worry about balloon and burst. So you certainly didn't need to worry about these things.

For the Westinghouse four-loop plan that we looked at, we did have higher peak cladding temperatures where you had ballooning and burst. But it was still a lot lower where you had to worry about these sorts of things. So as I had mentioned, whether these limits will ultimately be limiting, they very well may not be depending on what's predicted for fuel dispersal.

It's also hard to say because it depends on what the burnup of your core is, where you put the high burnup assemblies. If you only have lower burnup assemblies in the interior, then you don't necessarily have to worry about fuel dispersal and so on. And then maybe you start to push some of these other limits. But yeah, so we've done some calculations a while ago that look a little bit more like what we'd expect above the transition break size. And we haven't seen anything close to the embrittlement type limits.

CHAIR BALLINGER: There's a -- because of the speed of this having to be done, there's a WCAP which we're scheduled to see in a year. But we have and you folks have that takes the FSLOCA and expands the usefulness to all of these break sizes. And they don't get any -- they don't see anything.

MR. CORSON: Yeah, so I mean, to your point, this might be yesterday's problem.

MEMBER PETTI: That's exactly -- I said to you, I go, what the heck are we doing?

CHAIR BALLINGER: The train is moving. Okay. Believe it or not, we are actually ahead of schedule. We're scheduled for a break at 10:15. But the next presentation is likely to be at least as spirited as the one we're having now.

And so I would propose that we take our break now because we're going to have a longer presentation next. So it's now 9:38. Let's see if we can get ourselves behind schedule and reconvene at 10:00.

(Whereupon, the above-entitled matter went off the record at 9:38 a.m. and resumed at 10:00 a.m.)

CHAIR BALLINGER: Okay. We're back in session. It's 10:00 o'clock. So again, the floor is yours.

MR. CORSON: Okay. So what I talked about earlier this morning was all old stuff. And I can always blame other people for all the things that you didn't like about that. This is new, and I only have myself to blame for it. So here we go. So next slide, please.

You saw this exact same slide yesterday during Joe Messina's talk. I just want to remind everyone what we're talking about. We're talking about during high -- for high burnup fuel, experiments have shown that during under conditions that you would get in a loss of coolant accident, the fuel can fragment, relocate axially, and disperse into the coolant.

So the ballooning and burst stuff is driven by difference between the internal pressure, the cladding, and the reactor coolant system pressure. And so if you get this behavior, relocation to the balloon region can impact some things about peak cladding temperature and so on. Bursts, of course, can release fuel into the RCS which is not ideal, let's say. So next slide, please.

This slide is meant to show some of the history behind fuel fragmentation, relocation, and dispersal. It's meant to highlight that some of these things were understood quite some time ago, particularly the fragmentation and relocation part. And existing best estimate LOCA models often account for this axial relocation which can happen even for lower burnup fuel.

What changed around 2006 or so is that tests at Halden on very high burnup fuel around 90 gigawatt days per MTU. So far beyond what we have right now showed significant dispersal and significant fine fragmentation. And that's what really led us down this path to studying this issue more.

And so there's a few other sort of signposts along the way on this slide which I'll mention a little bit more in the rest of the presentation. So next slide, please. You also saw this slide yesterday during Joe Messina's talk. But basically, the whole point is that when the original ECCS criteria were established, this dispersal issue was not known.

The whole point of these rules or the whole point of the rule back then was to prevent a giant pile of rubble essentially forming because of embrittled cladding shattering from this rubble pile. When you start to talk about fuel dispersal, you're starting to approach that thing or that scenario that was the original rules were meant to avoid. So the current rules don't explicitly address this phenomena. Next slide, please.

So of course you heard yesterday we got direction from the Commission to address fuel dispersal as part of this rulemaking. So the rule language is a little ambiguous about how fuel dispersal will be treated. There's some statements of consideration in the Federal Register notice.

But a lot of the stuff is being left to guidance for how you address fuel dispersal. So I'll walk through what's in the guide. But at a high level, it includes a model to estimate how much fuel could be dispersed.

It provides some high level acceptance criteria for how much fuel you could disperse. And then it has a number of analyses to perform if you predict any fuel dispersal. So again, I'll walk through this in a lot more detail later on.

I'd also like to emphasize that this guide builds on a lot of recent research efforts, some at NRC, some through our international partners, some at the national labs. So I'd just like to highlight a few recent publications. So we had the Research Information Letter or RIL 2021-13. It was published at the end of 2021.

We briefed ACRS on this. So we've had some back and forth about this document. More recently, we sponsored this Phenomena Identification and Ranking Table or PIRT exercise related to fuel dispersal. So we had our first meetings on that about a year ago.

And then we published this report, NUREG/CR-7307. That documents the panel's discussions and conclusions. And so that was just published in May.

Separately, EPRI has been working on this, this white paper interpreting how fuel dispersal should be, how the research results should be interpreted. So we were aware of this. We've seen it.

And then there's a lot of work going on at the national labs doing additional testing as well as in the international community on this topic. So a lot of stuff going on. And we're using that to inform the guidance.

MEMBER PETTI: So just another question. This is harking back to yesterday. Philosophically, what changed that made you guys go from we really don't want any dismissal -- and that's sort of what the old rules implied, whether it was intended or not -- to, well, we're going to give you a little wiggle room, which almost to me, it sounds like a trap.

Because showing something doesn't happen, there's a good criteria, I think you can do it.

And it's a goal that has a lot of surety behind it, right? I mean, you can live to that. Once you open the door and allow fragmentation and you start asking questions about uncertainties, you're going to get into unknown unknowns. And you're going to be in what I call analysis hell.

You can't get your way out of it. Yet they provided in the guidance to acceptable -- equally acceptable ways. And I don't see it that way. I see them as really different. And to me, that's, again -- that's just not the way I think we should write regulations myself going forward. So any thoughts you guys have to help me kind of think through this.

MR. CORSON: Yeah, I mean, it's a really good question. I think part of the issue is that it's hard to say that you would get absolutely no dispersal at all, that it's zero. Because if you have ballooning and burst, even for lower burnup rods where we have fragmentation, it's much more coarse fragmentation.

But you have fragmentation. You've been at lower burnups. You could lose a very small amount of fuel from that. So certainly it's hard to show that it's absolutely zero unless you can complete prevent ballooning and burst.

Of course, at these lower burnups, almost nothing would come out. It's essentially zero, but it's not exactly zero. So at what point -- so we can also say, like, if a little bit comes out, if a few pellets' worth were to come out, let's say, that certainly seems like it should be pretty easy to cool.

And so philosophically, that should be acceptable. It may be very restrictive to say that you can't disperse any fuel whatsoever. But drawing that line as you say is very difficult.

I think maybe the guidance isn't written this way. But I think the staff would prefer to have no dispersal. That would make everyone's life a lot easier.

Philosophically, it makes a lot more sense. But we did provide this other pathway in the guidance. So I don't know. I guess it's a difficult philosophical question.

You can say that some amount of fuel you can probably show that it's coolable and doesn't have any other bad consequences. But as you say doing that is difficult and maybe a slippery slope and so on. So we're aware of this.

We've got direction from the Commission to address it. They didn't say how to address it. But we're trying to read the room, so to speak, on where industry is coming from, where we're coming from, and so on.

Come up with something that we find to be acceptable and practical. So next slide. So before I get into the guide, I think it helps to go through the conclusions from the fuel dispersal part. So this involved a number of experts from academia, from the national labs, from the international community who have studied either fuel dispersal itself who are experts in things like core design or thermal hydraulics.

So one of their big conclusions is that in order to demonstrate that the fuel would remain coolable, you need to know how much fuel comes out. And so for some of the parameters that impact this that they identify, you can calculate pretty well things like your core loading pattern. Obviously, that's going to affect how much high burnup fuel you have to begin with.

If it's in the interior versus the exterior is going to impact its temperature, whether or not it's vulnerable to dispersal and so on. So some of these things can be calculated pretty well. Other things can't be calculated so well.

There's high uncertainty for things like what the fragment size is, how much would come out of a fuel rod if it balloons and bursts, and so on. So they listed some of the things that are well known, some of the things that aren't well known that all impact how much fuel would be dispersed. One other big conclusion is that once fuel leaves the rod, it's pretty difficult to say where it will end up.

The thermohydraulics are very complex. The models are probably not quite good enough to predict with a lot of accuracy exactly where the fuel would end up. But what they said is that it should be possible to come up with some bounding scenarios, a few different scenarios where fuel could end up and demonstrate that the consequences would be acceptable.

So that was the PIRT panelist's opinion. But that's sort of our jumping off point for this guidance. So next slide. So this slide shows what the structure of the guide is, and I'll talk about each of these things in the following slides.

But it starts out with the limits on applicability for the models and the discussions in the guide. It talks about what the thresholds are for where you have to start worrying about fuel dispersal has the analytical limits for fuel dispersal. And here as I mentioned a little bit before, we have this sort of two options, no dispersal versus some dispersal.

If you have any dispersal, then you have to start worrying about the rest of the guide, being able to predict how much fuel disperses and then evaluating the impacts of that dispersal. So I'll go through all of this in the following slides. Next slide, please. I'll start with the limits on applicability.

So the test data that was reported in Research Information Letter 2021-13, it was all UO2 fuel in zirc cladding. So that's where we restricted the applicability on the method for estimating the amount of fuel dispersal. What I will say is that there have been other tests performed on fuel with dopants that could be used to support whether this would be applicable or not, whether these thresholds would be applicable or not to these dopants.

So unfortunately, a lot of this comes from protected data from, like, the SCIP-IV tests which were just completed in I think June. And there will be some additional upcoming tests. But the vendors have access to this data. We have our own access. So there is a way to extend this to different fuel types.

MEMBER MARTIN: But -- it's Bob Martin. Looking at your list here of examples, gadolinia seems to stand out, right? Everyone uses gadolinia, right? It would almost seem that the guide is to restrictive. Without gadolinia, it just puts a financial burden on applicants. Is there not enough data with gadolinia to draw similar conclusions or at least joint conclusions about undoped and at least O2 with gad?

MR. CORSON: There is, but it's not publicly available. So SCIP-IV did include some tests on gadolinia. As I said, the vendors have access to this.

Since the tests were just recently completed, we can petition Studsvik or the SCIP management board to release some of this data. Because of the timing of when this finished versus when we're developing this, we didn't have time to do that and get approval to consider whether we could include that to support this guide itself. So this is relying on what's publicly available right now. And so there is this current restriction.

MEMBER MARTIN: Is there nothing at Halden?

MR. CORSON: I don't -- they didn't do any LOCA -- they didn't do any LOCA tests on the gad fuel. So again, Studsvik, they did as part of SCIP-IV.

CHAIR BALLINGER: Do you have a feeling as to whether those results may affect this guide and/or will you likely have that information? This is a draft guide. We're going through this progression. By the time the draft final gets released for public comment, will that data be available for you guys?

MR. CORSON: So this data is already available to NRC. So I'll say I think we have enough information to say whether or not it applies to some of these different fuel dopants. But again, since this is a publicly available guide relying on publicly available information, to say, like, okay, this is applicable to a certain dopant based on some data that no one can see kind of says a little bit what that data has to say, so to speak.

So we have the data. We may be able to petition to release it more widely and update the guide to make it applicable or not as appropriate in the draft final rule stage. But right now, we didn't have the time to do that.

MEMBER MARTIN: You know, if it was one of those things you can look at the data and it's indistinguishable from undoped, might that be a slam dunk. I kind of say that there's enough differences that you would -- if you revise this in a few years because the data is available, you would probably have something that's a little bit different. And you have --

(Simultaneous speaking.)

MEMBER MARTIN: -- this little knowledge you may have.

MR. CORSON: I guess what I would just say is no comment at this point. We have enough information to make decisions on these dopants. The vendors do as well. Right now, we can't really talk about it so much in an open forum because it's protected as part of these projects that people pay into. And so certain people have certain expectations of what information can be shared. So --

CHAIR BALLINGER: But if there was a cliff, we would know it.

MR. CORSON: Yeah, we would know it.

CHAIR BALLINGER: To use a term that's been used around here, the cliff edge.

MR. CORSON: Yeah, we would know it. So yes, we're aware of this. Unfortunately, again, it's protected as part of the SCIP-IV project. We can release that information.

We petitioned the SCIP management board to release the SCIP-III data, that Research Information Letter. So there's a process to do it. We may be able to do it for the draft final rule, but we haven't gotten to that point yet.

MEMBER MARTIN: Appreciate that.

MR. CORSON: Okay. So yeah, that has to deal with the estimating the amount of dispersed fuel. The stuff later on about demonstrating coolability and so on, it's generally applicable to other fuel types. Of course if you had something pretty exotic like uranium nitride, there might be different things that you would have to consider.

But I think the principles are still pretty similar. So next slide. So the first thing that the guide talks about is the FFRD thresholds. And really the main concern in all of this is fine fragmentation.

It's not necessarily the fragmentation that happens at lower burnups. It's the really fine fragmentation. So you see on the left these figures that were in the research information letter that shows publicly available test data.

There's two figures, one just showing the fragments smaller than one millimeter, another showing smaller than two millimeters. So obviously there's more that are smaller than two millimeters. But the trends are pretty much the same.

And based on this data and some arguments in the Research Information Letter, we've concluded that fine fragmentation begins at a burnup of around 55 gigawatt days per MTU. We recognized that this is a real oversimplification of what's actually happening in the fuel. It's undergoing a lot of changes to its microstructure.

There's other things that impact these changes like temperature. All these things feed into fragmentation behavior. But to simplify things, we just set this burnup threshold in the guidance. So next slide.

MEMBER MARTIN: Just looking at it, the figure, you might say 60.

MR. CORSON: Yeah.

MEMBER MARTIN: Is there no data at 55? I mean, given all the tests that were done or just all hidden behind proprietary --

MR. CORSON: Yeah, so again, this is -- yeah, so one thing I'll say, this is what we released as part of the Research Information Letter in 2021.

And at that time, there weren't these types of tests in there. Obviously, anyone looks at this figure realizes, oh, yeah, we should probably do some tests in that range which we did as part of SCIP-IV which again is not publicly available right now.

MEMBER MARTIN: Thanks for that.

MR. CORSON: Yeah. I will say in the Research Information Letter, we did look at a test that was performed at Argonne National Lab. We didn't measure the size distribution. But qualitatively, it started to show some fine fragmentation at 55. So that was another argument in the Research Information Letter, another reason why we said 55 instead of 60 as you say where it jumps up.

MEMBER PALMTAG: This is Scott Palmtag. I have a question.

So, this says fragmentation begins at 55, current limits are 62 --

PARTICIPANT: Right, that's the question.

MEMBER PALMTAG: How does this work when these are optional?

MR. CORSON: Yes, so, you know, this is a really good question.

So, I'll talk about this in a little bit in another slide.

But one thing I will say is that, typically, the higher burnup fuel in current designs is on the periphery where it's not very vulnerable to ballooning and burst.

So, it's not as much of a concern.

So, I talked about a little earlier how we did these calculations ten years ago to look at fuel disbursal.

And for the Westinghouse plant, there was one high burnup fuel assembly in the center of the core, the very center of the core. Everything else was on the periphery.

So, all the stuff on the periphery never got hot enough to balloon and burst.

The one in the middle did and so, we had some estimates of how much fuel could be dispersed as a result of that.

So, this is, you know, obviously, an ongoing issue that we're trying to resolve.

Right now, we're focusing on the voluntary part of this rule.

But going forward, we'll have to see what needs to be done for the existing, you know, fleet under the existing requirements.

So, yes.

So, what I'll say also, you know, in 2015, we were -- well, before this, we were tasked by the Commission to consider fuel dispersal as part of 50.46c.

We did that. That was the reason we performed some of these calculations in the past.

We wrote a SECY paper to the Commission basically saying that, for various reasons, there is no need for immediate action.

And part of it is that, based on the way cores are designed, very little fuel that's acceptable could confine fragmentation and dispersal would balloon and burst during a LOCA.

It would only happen pretty late in cycle because you would have to get, you know, late in cycle to get to these burnup limits for some of these rods.

And so, based on some other, you know, qualitative risk insight type things, we decided there was no need for immediate action.

And so, we still believe that's true for existing fleet, existing core loading patterns and so on.

So, it's, you know, once you start getting to higher burnups that these issues are exacerbated and there's more of a concern.

And so, that's why we're focusing on it for this voluntary rule which is really meant for the higher burnups and enrichments.

Next slide?

Not only do you need, you know, a certain burnup or certain microstructural characteristics in the fuel to get fragmentation, the research has also shown that you need a certain amount of cladding strain.

So, you know, the cladding provides some restraint that prevents the sort of fragmentation behavior.

If you get enough strain, this is when you start to see this fragmentation and relocation.

So, based on a lot of data, we concluded that below about 3 percent hoop strain, you don't get this fragmentation and relocation.

And so, the figure on the left is just meant to how, you know, after one of these tests performed at Studsvik, this was a -- before the SCIP 3 program. This is an NRC sponsored test at Studsvik.

But after doing the test, they stuck a wire probe in to see how much, you know, how much empty space essentially there was. That's the kind of neon green line.

They broke the rod in half, shook out all the loose fragments, did the probe again to see how much loose fuel had come out basically, and then, did a gamma scan.

So, the black line shows the strain.

The blue is the gamma scan were you have a lot of strain. There's no fuel.

Where you have less strain, the fuel stayed there intact, everything else.

So, you know, based on all these tests, not just the NRC sponsored test at Studsvik, but the SCIP test, we concluded 3 percent strain is a reasonable threshold.

Next slide?

Okay, so, those are the -- that's meant to identify what rods would be susceptible to fragmentation and dispersal -- to find fragmentation and dispersal.

So, this guide provides acceptance criteria for dispersed fuel mass.

So, below the transition break size, we continue to say, no fuel dispersal.

And so, you address this by showing no ballooning and burst for rods with peak pellet burnup of greater than 55 gigawatt days per MTU.

So, that's, you know, below transition break size, straightforward.

CHAIR BALLINGER: Excuse me, this is Ron Ballinger.

We get confused with units. The 55 is pellet average burnup.

And now, you just used the word peak.

Are they the same?

MR. CORSON: Yes, yes, pellet --

CHAIR BALLINGER: Okay.

MR. CORSON: average, I should have said, yes. Sorry.

Yes, yes, there's a lot of burnup measurements.

It's just meant to say like if you take a, you know, pellet or an axial slice, that's what the burnup of that actual slice is.

It's not meant to say, you know, the radial local burnup distribution of anything like that, it's, yes, this axial slice is what we're talking about.

So, for breaks above the transition break size, as I mentioned, you know, there are two options.

No dispersal, which is maybe, I don't know, saying the easy way is really the right terminology because you need to design your core in order to show this, but it's certainly the more straightforward way. You've just shown no dispersal and you don't have to worry about all these other things.

Or, you know, as I'll talk about on the upcoming slides, if you predict some dispersal, there are some additional considerations.

So, the dispersed fuel mass should be calculated using an approved valuation model and the field dispersal models.

So, the fuel dispersal models, that's what's in this guide.

The approved evaluation model that's, you know, based on existing guidance and whatever new methods the vendors come up with and propose to us to deal with above the transition break size or to do these new sorts of analyses.

So, we did not go into all the details of developing or updating the guidance to say how the evaluation model, part of it would change to do these sorts of analyses.

One other thing we did specify, sorry, is to say that the evaluation model should consider the impact of transient fission gas release.

Part of these tests at Halden and Studsvik showed that you can get additional releases of fission gas in part due to either microcracking or this fragmentation of the fuel under loss of coolant accident conditions.

Next slide?

MEMBER MARTIN: Going back there just a little bit, isn't it on the previous slide were you talk about the 3 percent strain limit and then, talk about it in context of it's backed up by data.

Ultimately, the data has limitations, right? And there's, you know, you've looked at burnup only so far.

You looked at all these certain fuels, et cetera, et cetera.

It seems like it's missing that qualification, like, you know, we look at fuel burnup up to 80 or whatever, whatever the data.

You don't just leave the door open, you know, even though people probably aren't going to go, yes, all the way.

But, you know, you defend it with data, you've got to qualify limits with the limits of the data set.

MR. CORSON: Yes, no, that's a good point.

I will say, you know, for the Halden test, the tests were done up to I think as high as like 92 or 93, I think it was perhaps used, so very, very high burnup.

SCIP, I don't recall. When I have my slides up, I'll be able to --

MEMBER MARTIN: Well, not every test may be measured --

MR. CORSON: Yes.

MEMBER MARTIN: -- at this rate either.

MR. CORSON: Yes, so, the Halden test, I think that that's an issue. They didn't necessary measure all the strain.

For the SCIP test, they went up to closer to like 80 gigawatt days per MTU.

So, that's maybe getting closer to where we would expect.

MEMBER MARTIN: And they measured the strain?

MR. CORSON: Yes.

Yes, so for all of the SCIP 3 tests, and for the NRC sponsored test at Studsvik, they measured the strain. They did a lot more detailed measurements.

In the end, Halden did a lot of detailed measurements, but back in, you know, 2006 when they first discovered this widespread dispersal, they didn't do nearly as many as detailed measurement as we did subsequently.

But yes, up to I'd say like 80 we've done a lot of these more detailed measurements.

MEMBER PALMTAG: This is Scott Palmtag again.

I'm going -- Ron, you brought up a good point about units because we talk about 62, that's rod average.

MR. CORSON: Yes.

MEMBER PALMTAG: But your local burnup could be 70 on a rod average.

And then, you're -- but you're saying the fuel fragmentation limit is a local burnup, not quite the peak, but the local burnup or maybe a couple pellets?

MR. CORSON: Yes, yes, that's the case.

So, one thing I'll say about this is, what we've seen in a lot of our calculations, and I think this is consistent with other calculations that we've seen, the first location tends to be pretty high up in the rod where you're maybe a little bit closer to what the rod average is than you might be at another location in the rod.

But yes, this is local phenomena.

And I'll say the calculations we did earlier and the conclusions we made earlier based on whether we needed to do anything for the current burnup limits and so on, that was based on this understanding of peak local behavior of what the pellets would be at a certain location.

So, even with that in our calculations, we still then see, you know, lots of fuel dispersal for existing types of core loading patterns.

CHAIR BALLINGER: So, later on in the guide, you probably have looked at the slides, that when it comes to determining how -- dispersal from where?

You're using between -- you're assuming it's between spacer grids, right?

MR. CORSON: Yes.

CHAIR BALLINGER: So, high up in the rod it's a big gradient between spacer grids.

And so, that's a little bit of a -- you're having to assume that the releases for, I don't know, ten inches or whatever the number is --

MR. CORSON: Yes, I mean --

CHAIR BALLINGER: -- between spacer grids.

MR. CORSON: -- that would be one way of doing it if you had a more detailed model that showed, you know, strain is more limited within that spacer grid span then you could account for that.

But yes, at a minimum or, I guess, at a maximum, it would be limited to the distance between spacers.

Just the reason we said this in the guide is that, you know, most codes, you're not going to account for exactly how the spacers impact the strain and all of that.

So, it's just easier to say, okay, we're going to cut it off.

Like when we do calculations with our FAST code, we don't account for the spacer. So, we might see like several feet above 3 percent strain.

But of course, the spacer grids are going to more realistically restrict what that strain is.

CHAIR BALLINGER: Just because there's -- in general, I don't know if it's codified, there's a 1 percent strain limit for design on the cladding.

MR. CORSON: Yes, that -- I mean, that's for, you know, like normal operations --

CHAIR BALLINGER: That's the starting point.

MR. CORSON: Yes.

MEMBER PALMTAG: Just go back to limits, but you're saying that the limits that you've seen have always been on the outer edge of the core? Not putting words in your mouth.

MR. CORSON: Sorry, say that again?

MEMBER PALMTAG: You've only seen where the 55's exceeded, has it always been assemblies on the outer edge?

MR. CORSON: Yes.

(Simultaneous Speaking.)

MR. CORSON: So, yes, that's what we see from most just in core designs.

I don't know, John Lehning might be able to say more about this because he deals more closely with this stuff.

MR. LEHNING: This is John Lehning from the staff.

I would just say that there's not any restriction or requirement that, in general, that fuel above that 55 gigawatt day peak or pellet threshold has to be on the periphery.

And so, I think what James is talking about is certain typical designs.

And so, sometimes, you will see that those higher burnup assemblies will be on the periphery, but there's not some restriction.

And I guess the other thing I would add that I think Member Palmtag was not present for our review of a topical report that dealt with this issue that was presented maybe before the ACRS, WCAP-18446.

And there was some consideration and some discussion then where we went into some more detail about the issue of current operating fleet and that burnup range.

And the staff presented some additional thoughts about how we're continuing to look at that.

It was also a non-concurrence there about whether that was being dealt with in a sufficient way.

But there is more information there. I won't go into a lot more detail on this comment, but certainly, we could explain a little bit more about that if needed.

MR. PALMTAG: I just - I mean, I agree with you that if the burnup is limited to the outer edge, we'd have much lower probability of ballooning and bursting, the power is going to be very low.

If it starts getting further inborn, that may not hold up.

CHAIR BALLINGER: Well, if they go to a

24-month cycle and increase enrichment to 7 or 8 percent, then they start pushing these bundles in -- further into the core in high burnup.

MEMBER PALMTAG: I don't think so.

CHAIR BALLINGER: You don't think so?

MEMBER PALMTAG: I think they'd still be on the outer edge, but I mean --

CHAIR BALLINGER: I mean, outer edge.

MEMBER PALMTAG: Yes.

MR. CORSON: Yes, well, it'll depend on the core design. This is something for vendors to figure out.

What I will say is that the DOE NEAMS program worked with one utility and Westinghouse to come up with, you know, just a hypothetical core design that is 24-month cycles, goes to high burnup, higher enrichment.

And for that core design, they did have high burnup fuel in the interior.

We did our own calculations using this core design just to sort of demonstrate our methods for doing calculations of fuel dispersal and, yes, there were a lot of high burnup assemblies at end of cycle that were above this burnup threshold, quite a bit above.

But that was a hypothetical core design. It was just, you know, an academic exercise.

What vendors would actually come up with remains to be seen.

Okay, so, we've been talking about this a little bit, but I guess I'll address it right now in this slide.

The plausibility of this no dispersal criterion.

I talked about these analyses that we did about ten years ago on the different plant designs.

Based on current license burnup limits and fuel management practices, so again, where we have primarily the high burnup assemblies on the outer ring of the core, using generally nominal initial conditions, maybe not quite as far as what we're anticipating for this rule above the transition break size, but certainly closer to that than what you would get in licensing methodologies.

And what we saw for those calculations for both large break and some small breaks that we looked at is that we didn't get any dispersal for the CE PWR or the GE BWR for -- that we looked at.

The peak cladding temperatures were just too low.

The CE plant was getting, you know, close to where you might start getting some ballooning and burst, but the BWRs was just far below areas where that would be a concern.

So, these were just three specific calculations, plant types. It doesn't necessarily mean anything for the entire fleet, but it does show that there are certainly certain plant designs where this is plausible.

So, the other thing I'll say is that we specifically asked this question during the public comment period on the regulatory basis.

We asked industry, would it be -- do you think it would be plausible to show no dispersal using more realistic LOCA methods?

And generally, they said, yes, we do think that this would be possible.

So, we, you know, took their word that they think using their own methods, they can do this. And we went with it.

So, certainly, you know, this would be the more straightforward path showing no dispersal.

So, next slide?

But if you don't take that straightforward path, things get a lot more complicated.

So, I'll start with a method that we had for estimating the dispersed fuel mass.

So, I alluded to this a little bit when I talked about the fuel dispersal part. But there are a lot of things that go into how much fuel disperses, not just the fragment size distribution, but things like how big your burst opening size is or what your rod internal pressure is.

And so, this on the left, this is another figure from the research information letter contrasting two tests that had, let's say, similar fragmentation profiles, not exactly the same.

Obviously, the one on the bottom had more of the big fragments.

But they also had very different burst opening sizes.

And you can see the one with the very large burst opening dispersed a lot more fuel during the test, even for some of the larger sizes.

Whereas, you know, the smaller burst opening, as you would expect, not as much comes out.

So, there's a lot of really complicated stuff going on.

A lot of these processes are highly stochastic, not really well understood.

So, rather than coming up with some mechanistic way of estimating fuel dispersal, we had a much simpler surrogate model that was based on something on the research information letter.

MEMBER ROBERTS: Real quick, if you would explain the difference between dispersed and collected mass?

MR. CORSON: Yes.

So, I talked -- I mentioned this very briefly when I was talking about the relocation stuff.

But basically, the test procedure is you run this simulated LOCA test.

It balloons and bursts, some fuel comes out.

You collect that and measure it.

That's the dispersed mass. So, that's the blue in these things.

What is then done is -- or what was then done was the rod was broken in half and shaken to shake out loose fragments.

So, obviously, this is not something that happens in a LOCA.

But it was just meant to try to quantify how much loose stuff there was in the rod.

And so, you can see, you know, on the

bottom, there's a relatively large amount of loose stuff in the rod, but it all stayed in there until we broke it apart and shook it out.

Whereas, on the top, a lot more came out initially.

MEMBER HARRINGTON: This is Craig Harrington.

So, the fragmentation or the fracturing of the pellets happens, but without the strain the pellet basically stays intact.

Is that --

MR. CORSON: So, the --

(Simultaneous Speaking.)

MEMBER HARRINGTON: -- understanding?

MR. CORSON: The pellet pretty much just stays intact when you don't get a lot of strain. You need a lot of strain for it to fragment, you know.

And there's a lot of complex processes exactly what's leading to the fragmentation isn't fully understood. It's still debated.

But yes, there's, you know, a number things that are happening. And it does rely on a certain amount of strain for this to happen.

MEMBER HARRINGTON: And is that a distinction between fracturing and fragmentation?

MR. CORSON: Yes, yes.

So, like normally, during normal operations, the pellets do fracture. I mean, they undergo a lot of thermal stresses. They fracture. So, they're -- you can look at some pictures, micrographs that you see all these big cracks in pellets.

But they're big cracks. So, they're essentially still intact pellets.

It's during the LOCA that you get these basically rubble or sand at really high burnups.

So, yes, that's the distinction.

MEMBER MARTIN: One follow up on that collected.

You're cutting a segment and then, shaking it out or --

MR. CORSON: So, the whole -- so, these tests are performed on segmented rods.

So, you take a full length rod and you cut it into pieces.

MEMBER MARTIN: How long?

MR. CORSON: For the Studsvik test, they were I think up to 30 centimeters, so a foot --

MEMBER MARTIN: Okay.

MR. CORSON: -- basically, so pretty

small.

Halden, they went a little bit bigger, but I think no more than 50 centimeters, so a foot and a half roughly.

So, yes, they're pretty small samples.

You know, there's a lot of discussion about how this relates to a real loss of coolant accident.

You have higher gradients in this furnace because you're talking about a pretty small sample. How does that affect the ballooning and burst behavior?

So, yes, there's a lot of really complicated stuff. How well this represents real life is still hotly being debated right now.

MEMBER MARTIN: Yes, we're seeing that the -- these results with the collected mass would be sensitive to that segmentation.

MR. CORSON: Yes.

So, I think that's why we tried to also define things in terms of burnup and strain thresholds that you calculate rather than just correlating how much mass comes out because of this length effect.

So, I think our philosophy was, it's easier to, you know, you don't know exactly the extent of the balloon and so on, so it's easier to correlate to something like the --

MEMBER MARTIN: Yes, sure, that makes more sense, for sure.

MR. CORSON: You know, the collected mass just seems like something you could take with a grain of salt, and the pun was intended. Right?

Yes, I mean, some of these probably are that size, so, yes, it's appropriate.

MEMBER HARRINGTON: Just to help explain -- I'm Craig.

But my understanding is it's the pellet can fragment at any time, but you still have the cladding contact, so it's still holding the pellet together.

MR. CORSON: Yes.

MEMBER HARRINGTON: It's not to the balloon and burst that it gets released again.

MR. CORSON: Yes, so, yes, there's a lot of processes that lead to this fragmentation or fine fragmentation.

The fracturing, that's something that happens at normal operations. And as you say, the cladding holds that all intact.

Okay, next slide?

All right, I apologize for how small the font is on here, but this is, you know, figures from the research information letter that you can go back and look at.

But what it's showing, the left most figure, this is our correlation showing the fragments that are smaller than one millimeter.

And I'll just say, we just drew a line to try to include most of the points. We didn't do any sort of statistical analysis or anything like that.

This was just a pretty simple line meant to bound most of the tests showing fine fragmentation behavior.

MEMBER MARTIN: This is the same plot as before with a line on it?

MR. CORSON: Yes.

MEMBER MARTIN: Okay.

CHAIR BALLINGER: There's a lot for doing that rather than some sophisticated multi-parameters to physical fit within R square or point one?

MR. CORSON: I mean, I'm not that smart. So, I was like, you know, this line looks pretty good, so, yes, that's what I did, you know, a few years ago and it seemed to make some sense.

So, yes, on the other figure it shows the actual collected dispersed mass during the test.

And you see this similar sort of trend where, you know, lower burnups, you have no dispersal or very little.

Higher burnups, you get more.

So, we had this line showing small fragments basically.

We have this other curve with a similar shape showing how much gets dispersed.

So, we -- I'll let you ask your question.

CHAIR BALLINGER: Is there any correlation with the rim?

MR. CORSON: Yes, so, as I said, the exact mechanisms are not fully understood. But certainly the high burnup rim contributes to this a little bit.

But it's probably not just the high burnup rim, there's also some restructuring where you're moving towards the rim a little bit more into the pellet where fragments seem to be coming from that as well.

Again, the exact mechanisms, what's causing it, not fully understood, but certainly the rim -- high burnup rim plays an effect right here.

And so, there is some discussion about like, you know, at what burnup do you start to form a rim?

Does it start to get thick? And those other considerations.

It's -- you start for form it before 55, but it's, you know, really, really thin then.

So, yes.

MEMBER HARRINGTON: This is Craig again.

Is there anything unique about the one very low data point?

MR. CORSON: Sorry, which particular data point?

MEMBER HARRINGTON: Forty-two, 43 gigawatt days?

MR. CORSON: Yes, I mean, that was just, you know, based on some -- the testing that we showed in the research information letter that was like the low burnup sample that we had.

Certainly, there have been a lot more tests that have been conducted on fuel at much lower burnups than that in various international programs.

In all of these programs in the past, we never saw any sort of significant fuel dispersal.

So, this wasn't something that was measured going back to those older LOCA tests and Phebus, for example, in France or PBF.

But certainly, if you would keep extending this, you would see, you know, no dispersal, basically no fine fragmentation for those lower burnups.

So, this is just like the data set that we chose to show in the realm. There is this one point at this burnup that was in the SCIP 3 program.

MEMBER HARRINGTON: So, you just have a range of things that some of which had very low, no mass fraction release even at high burnups over and across the spectrum?

MR. CORSON: Yes, you know, I said earlier, burnup is just one thing. There's a lot of other parameters that go into it.

And you can, like you say, there's a lot of scatter in this data and there's certainly some, like you said, there's some tests that show less fine fragmentation.

I think you can look at this and you can see a lot of those low points are from Halden which, you know, Halden, the tests were performed in a reactor with some internal heating.

They also supplemented it with furnaces to get the cladding temperatures they wanted.

But those tests, in some ways, were different than what's in the, you know, performed in the hot cell.

So, what difference does that make? You know, is that important? Is this just a coincidence?

I'll say we have -- I talked about Oak Ridge is doing a lot of tests in their hot cell on samples from Byron, for example, in the U.S.

There's also going to be tests at Oak Ridge -- or sorry, at Idaho National Lab in TREAT.

So, then, we will have an in reactor internal heating test to see, you know, does this change the fragmentation behavior.

But yes, right now, this is the information we have.

There's a lot of scatter.

You could argue that our -- you know, the curve we have is conservative, which it probably is, but we're talking about the situation where you disperse fuel with a lot of unknowns and other situations.

So, it's maybe appropriate to be -- to err on the side of caution there.

MEMBER HARRINGTON: Has anyone just looked at burned fuel and just open up and see if it's fragmented or any correlation on that?

MR. CORSON: You mean, just normal

operation?

MEMBER HARRINGTON: Yes, normal operation?

MR. CORSON: Yes, I -- for all these tests, what's simply done is they do like a before. So, they take a sample that they're not going to test and take a slide and do some optical microscopy or some other SEM, whatever else.

And they look at the micro structure before the test and a sample, maybe slightly above where they're actually testing.

And then, they look at, you know, similar microscopy, SEM after the test.

And yes, that -- you don't see any of this fragmentation happening at high burnup fuel from in reactor stuff.

Again, you get some fracturing, you don't get fragmentation until you put it through loss of coolant accident conditions, get significant strain, and so on.

So, this is a -- really a LOCA phenomena -- phenomenon and it could potentially happen in some things that have similar heating transients to a LOCA, but --

MEMBER PALMTAG: So, the LOCA's actually what causes the fragmentation?

MR. CORSON: Yes, yes.

MEMBER PALMTAG: Even if it's no burst?

MR. CORSON: Yes.

So, if you did have enough strain but didn't burst, so there is also some question about this exactly when you get fragmentation.

Typically, what we do, we have, you know, we run the tests. We know what it looks like before. We know what it looks like after.

We don't know exactly what's happening at each point in time, when exactly it's fragmenting.

But certainly, we think that if it balloons and doesn't burst, you get fragmentation.

And there was one Halden test that was specifically designed to balloon and not burst. We still saw fragmentation behavior.

But as you saw in the curve, those Halden tests had less fine fragmentation. It's fragmented, but they were bigger pieces basically.

MEMBER MARTIN: James, it's Bob, again.

When you mention LOCA, it's some other events that look like, I assume, and includes reactivity insertion, accidents of same likelihood in that design basis realm?

Nodding you head.

MR. CORSON: Yes, so, this is --

Okay, so, right now, we really focused on LOCA to try to make our lives a little bit easier.

The PIRT did look at different scenarios including reactivity initiated accident.

So, what I'll say bout reactivity initiated accidents are you need a specific set of circumstances to lead to this sort of behavior.

So, you would need something that would not fail the cladding due to pellet cladding mechanical interaction, but would fail it due to ballooning and burst, essentially.

And so, you need the right pressure differential across the cladding.

You need the right temperature transient and so on.

Those conditions are pretty unlikely and I think the -- like much less likely than getting this sort of behavior in a loss of coolant accident.

So, the panel -- PIRT panel concluded it wasn't really much of a concern.

I will say that we have guidance for reactivity initiated accidents, separate guide already in use that has limits to prevent violent expulsion of fuel.

There, we're talking about higher enthalpy where you hit the rod with enough energy to basically blow it up.

You know, going back to some of the SPERT type --

MEMBER MARTIN: And that's been around a while.

MR. CORSON: Yes, it's been around a while.

So, that limit's been there.

Whether you know you would need a lower enthalpy target for higher burnup fuel to prevent ballooning and burst in these situations, I, you know, it's still somewhat of an open question.

We're pretty comfortable that it's not a big issue.

MEMBER MARTIN: Really did no new tests looking at that in that context to identify an enthalpy limit.

MR. CORSON: So, another program that we participated in at NRC is the Cabri International Project.

So, the Cabri reactor in France, it's doing tests in a pressurized water reactor loop where they might start to get conditions closer to this where you would have to worry about fuel dispersal.

So, we're part of that project. The tests are still ongoing and so, it remains to be seen.

I'll also say, you know, TREAT is also performing tests on high burnup fuel as part of the FIDES project that we're a part of.

And so, again, that's another opportunity to make sure that there's no issues that we discover in these tests on reactivity insertion accidents.

MEMBER MARTIN: I think the technical basis for that enthalpy limit also goes back to some Japanese tests, too.

MR. CORSON: Yes, we --

MEMBER MARTIN: And I can't remember the name of the --

MR. CORSON: NSRR.

MEMBER MARTIN: There you go, thank you.

MR. CORSON: Yes, actually, most of the -- I'd say most of our data for RIA probably comes from NSRR. They've done a lot of really good, interesting tests.

So, that forms a lot of the basis for our guidance on RIA.

I see Dennis has his hand up?

DR. BLEY: Oh, yes, thank you.

Dennis Bley.

It's a very simple question, very simple question for you.

As far as we know, this is strictly a physical process it sounds like. And the bits coming out are, I assume, chemically inert ceramic bits.

So, we don't have any kind of weird chemistry we're worrying about peak and the rods falling apart, right?

MR. CORSON: Yes, I'd say, you know, yes, they're pretty stable.

I mean, I can't say for very long-term whether there would be -- if you had dispersal like if we're talking days, weeks, so on, if there would be significant issues.

You know, there's already this -- these sorts of considerations for leaking rods and reaction rates with UO2.

But, yes, as far as like immediately what happens, it's physical phenomenon.

There is some question about whether moisture does impact the dispersal, whether or not the particles are sticky, for lack of a better word, and whether that would impact how much fuel comes out.

So, it's still, you know, somewhat of an open question. It's sort of physical, sort of chemistry, surface effect type things, still unknown.

But yes, it terms of, you know, like long-term stuff, UO2, water interactions, it's not as much of a concern.

DR. BLEY: Okay.

And, yes, I know, we just don't want this stuff out there, but no real concerns about where it could accumulate, if it would accumulate. And it doesn't sound like it's likely unless these surface effects stuff.

MR. CORSON: Yes, I think, you know, this is what Member Petti was getting at.

There's -- once you start allowing dispersal, there's a whole lot of things you need to start considering.

And some of them, we do specifically have in the guide and have some methods to deal with, maybe just high level conceptual methods, but nevertheless.

Then, there are other things, longer term things, equipment qualification, or environmental qualification of equipment, dose effects, all these sorts of things which this guide doesn't deal with.

To some extent, like, you know, I mentioned environmental qualification and dose, you'll hear about Reg Guide 1.183 later and how it thinks about some of these sorts of things.

But yes, you know, long-term, whether there's chemical effects, it is a question that would need to be considered if you go down this path.

Okay, so, just getting back to this slide, what I'll just say is that, you know, we provided this model for dispersal.

The model is based on the small fragments but it doesn't mean that all small fragments disperse or that all that disperses is just fragments.

As, you know, I showed on the previous slide, you can get some larger fragments dispersing, some smaller stuff can stay in the rod.

But this is just like a surrogate for all the complex things that go into field dispersal.

So, next slide?

And this just compares this -- if you would apply this surrogate model to these tests that were performed at SCIP 3, what would you get?

And in most cases, you would over predict how much fuel was actually dispersed during the test.

So, that's how much came out and was collected before any short of shaking or anything like that.

In most cases, we're over predicting that.

In some cases, we are slightly under predicting that.

So, it's not necessarily conservative in all cases against this test set, but it is, you know, meant to be, you know, somewhat realistic, perhaps erring on the conservative side.

But yes, not intentionally conservative.

And we mentioned this earlier, it has this on here, the guide also says you can credit spacer grids to limit the axial length of fuel dispersal.

And there's a number of things that the spacers do that all act to sort of limit how much stain you get. And that's why we took this position.

So, if you predict high burnup rods that are above the fragmentation threshold, if you predict them to balloon and burst, you should perform this sort of calculation to estimate how much fuel comes out of those rods.

So, next slide?

MEMBER MARTIN: I'll pick on you again.

Going back to my comment earlier about, you know, the limitations on limits that are established from data, you know of mentioned the 80.

But here, you have last mass friction, you know, a burnup greater than 80.

Why don't you just say you stop at 80 because that's where the data stops?

MR. CORSON: Well, I --

MEMBER MARTIN: I mean, it seems to imply that you can support beyond 80, but you don't have the data to support beyond 80, well, everything.

MR. CORSON: Everything comes out after 80, so that's like the --

Again, we had -- we have the Halden tests that were above this 80 gigawatt of days where they didn't -- they didn't do this rigorous measurement of exactly what came out.

But qualitatively, a lot came out.

So, to say --

MEMBER MARTIN: Well, I'm just saying across the board going to like the 3 percent comment I had before.

You kind of said, well, the data only goes up to 80, you know, it just gives the impression that, you know, that Reg Guide can apply above 80 when maybe you don't have everything.

MR. CORSON: Yes, that's a good point and that's maybe a limitation that we can add that, you know, we think it's -- or we're confident it's only good up to about 80 gigawatt days. If you go beyond that, you need to provide additional justification.

So, I think that's a good point.

Okay, next slide?

So, I mentioned this earlier, but, you know, we showed in the RIL that transient fission gas release is an important thing to consider.

And what we're concerned about is that there's a situation where you might not predict ballooning and burst without accounting for transient fission gas release.

But if you account for it, it would lead to the rod ballooning and burst.

So, what -- one very important caveat I have to the figure on the left is that most of these tests were conducted at peak temperatures that were a 1,000 degree Celsius or 1,200 degrees Celsius.

So, you would already be ballooning and burst before that.

So, these tests, some of them did measure the release with time.

Others, we just had, at the end of the test, this is how much is released.

So, certainly, we would expect the extent of transient fission gas released to be less than the highest points on this curve.

But there is enough and to show, it may be, you know, 5 percent or so, and it is impacted by burnup.

So, we didn't endorse the model in this guide, but we did point to some papers in the literature that discuss this, that discuss temperature effects, and so on that could, you know, sort of get people on the right track, so to speak.

Next slide, please?

So, now, we're getting into dealing with the consequences of that dispersal.

So, the first part deals with transport of the particles in the reactor coolant system.

The PIRT had noted that the transport of these irregularly shaped particles that we expect from fragmented uranium dioxide in multi-phase flow is poorly understood.

So, to come up with a model for tracking where all of this goes would be very difficult.

You could potentially perform some tests to get some qualitative estimates and there is some work, I think, going on right now in industry and at EPRI related to this.

But yes, it's really difficult to track

where things go.

So, what they suggested doing, the panel -- the PIRT panelists, is they suggested coming up with some more simplified scenarios for where the fuel ends up and analyzing those.

So, that was the approach that we took.

So, one of the most likely scenarios they identified was that material would get trapped on spacer grids. So, certainly, spacer grids can collect some of this material, particularly the bigger particles.

So, we recognize that not all material would get trapped on spacer grids. But if you're going to come up with a bounding problem, that seems like a good place to start.

Certainly, as you'll see later, you know, some particles could fall down to the lower head or lower regions. And so, we have ways of addressing that as well.

The next slide, please?

So, assuming, you know, certain configurations, we have a few considerations for how you address fuel coolability.

So, first of all, we -- the guide says that you should address the impact and dispersal on the peak cladding temperature and maximum local oxidation.

I'll note that we don't have details in this guide for how that would be done. So, I think that's a shortcoming for how you would consider the impact of dispersed fuel on PCT and maximum local oxidation.

What I will say is that we're -- we have plans internally to do some calculations to look at this type of thing in the next couple months or next few months.

So, hopefully, when it comes time to the draft rule stage, you might have a few more thoughts on this matter.

On top of that, one of the issues with the particle bed is that, even once the particles might be cool, if there's internal heating, if the bed porosity and particle size is -- has certain characteristics, you could reach a point where the water is essentially pushed out of the bed and it can't remain cool anymore.

And so, this -- there have been a lot of tests looking at more severe accident scenarios to look at this sort of thing.

There's this 0-D Lipinski model that was developed based on these sorts of tests.

So, we say, you know, this is a model that can be used to look at, you know, these debris beds. And it has some ranges of conditions to consider.

And the particle size range is just based on the available test data that we have looking at different particle sizes from the tests.

And then, the bed porosity also based on some measurements of the porosity within the fuel rod so it's, you know, not quite what you would get outside the fuel rod, but it's informative nevertheless.

So, next slide?

So this just show, you know, the Lipinski model. These plots come from OECD/NEA Working Group on Fuel Safety report that looked at FFRD.

And so, I believe it was some German colleagues did these calculations for different debris bed heights.

They did show that under, you know, certain conditions, if you had really small particle sizes, high decay heat loads, you could reach these dryout heat fluxes.

So, it is something that would need to be verified if you did have fuel dispersal.

So, next slide?

So, the guide also says a few things about long-term cooling.

One of the issues with long-term cooling is that you could get, you know, some deposition or some chemical precipitates within the bed that could change its porosity.

And so, you know, that could impact the dryout heat flux.

Of course, in these longer time frames, the decay heat goes down a lot, too.

But it's something that would need to be considered.

Next slide?

MEMBER PALMTAG: I just had a quick --

MR. CORSON: Oh --

MEMBER PALMTAG: Sorry, this is Scott Palmtag.

I just had a quick question.

You're talking about these beds that would be hard to cool.

How much fuel would have to fail before you'd get -- and I'm thinking a few rods would be negligible.

MR. CORSON: Yes.

MEMBER PALMTAG: But it needs to be a lot of fuel to fail to get a bed this big.

MR. CORSON: So, it would have to be -- so, within an assembly, if you fail, let's say, most of the rods because you're at -- you have a high burnup of assembly, you know, inter rod peaking factors are, you know, there's some differences, but you're going to get similar behavior within a given bundle.

If all the rods were to fail within a bundle, you could start to get these debris bed heights of, let's say, ten centimeters, which was used in some of the working group on fuel safety calculations.

You can start to get those sorts of debris bed heights.

So, it is possible where --

MEMBER PALMTAG: I'm trying to understand that.

So, every fuel assembly would fail --

MR. CORSON: So --

MEMBER PALMTAG: -- but in the regions between the fuel grids?

MR. CORSON: So, it would be just within a certain assembly.

Basically, if the fuel is going to collect on a spacer, it's -- the fuel's not going to move necessarily a lot laterally.

So, it's most likely going to stay within the assembly that it came from or it's going to go up and down or disperse outward.

But so, within one assembly, if the rods fail within that assembly and fuel doesn't relocate, you know, radially, you could start to get these larger debris bed heights, just considering what's between two spacer grids.

MEMBER PALMTAG: Okay.

MR. CORSON: So, next slide?

All right, so I think the last set of analyses that I talked about or the last set of consequence considerations I talk about is this possibility for recriticality of dispersed fuel.

So, the PIRT panel did consider this a little bit. Basically, they said it wasn't much of a concern and that there are existing tools that could be used to pretty much do away with this concern.

And so, we did some calculations to do this.

What we did was we came up with a simplified model of the lower plenum. We just did a wet Westinghouse four loop, but this should be applicable in principle to other designs.

We assumed that all the fuel that dispersed was at its most reactive configuration, so that's the lowest burnup where you start to get dispersal.

And that it had an initial U-235 enrichment of 8 weight percent.

So, you know, pretty high -- pretty reactive conditions.

We didn't credit soluble boron, either.

So, next slide, please?

So, basically, this is a really extreme, extremely conservative scenario.

And when we did these calculations and calculated the effective multiplication factor, for very large masses of dispersed fuel, we were still far below any scenario where you would get at criticality.

So, yes, this just has some context for how much fuel in a really extreme case could disperse where if you had, you know, half of your rods were at high burnup, they all failed. Ten percent came out of each rod, extremely conservative, way beyond what's in our models, that would still be less than five metric tons of U-235 -- or sorry, U-02.

And so, we still don't even get any sort of criticality concerns far beyond that dispersed mass.

So, next slide, please?

So, the analysis showed This is very unlikely.

There were some earlier calculations that were in this same working group of fuel safety report. Those calculations used a lower enrichment which is why we went back and looked at this again to make sure that higher enrichments aren't a concern.

We did only look at this one configuration, but based on some other engineering judgments and how far we are from any criticality concern, we concluded that it's really not an issue that would need to be addressed.

So, next slide?

I think this brings me to the conclusions.

So, this guide provides guidance for addressing the impact of fuel dispersal.

There's a lot more detailed stuff or a lot more surety of what needs to be done to show no burst. If you -- no burst for high burnup rods, that is.

If you start to disperse fuel, there's less clear guidance, more high level considerations because that's, you know, not an ideal path to go.

And as I will stress with any guidance, this is just one method of meeting the regulatory requirement. Vendors can, you know, of course, propose something else.

So, that's all that I have.

Any questions?

MEMBER PALMTAG: So, kind of a big picture here, but you have failure, you have disbursement, you get a lot of it, 20 centimeters on the bottom of the beds.

So, what's the limit? I mean, what would say is too much disbursement?

MR. CORSON: So, I think, you know, that -- it's really hard to say. That 20 centimeter bed, I think the calculated mass was like 2 metric tons. That's a lot.

I would still say that's a lot and not desirable.

For some of our calculations that we've done and that industry have done for This sort of hypothetical core loading pattern with lots of high burnup fuel in the interior, a lot of other issues with it that would cause it to not be a realistic loading pattern.

But you've been with that kind of unrealistic loading pattern, we're getting more like 400 kilograms of dispersal or less, depending on what assumptions you go into it.

Whether that would be acceptable based on, you know, the considerations in the guide, we, unfortunately, didn't have a lot of time to do these sorts of calculations.

A few hundred kilograms, I did some quick calculations. I think 100 kilograms of fuel in like 80 percent packing fraction is something like 11 or 12 liters for 100 kilograms.

So, I mean, that's a decent amount of fuel for just a 100 kilograms.

Is that acceptable or not? That's something we're, I guess, struggling with really, to be honest.

MEMBER PALMTAG: I guess that's my question, what's the limit?

MR. CORSON: Yes, I mean, the simplest in the guide is zero or come up with something better.

So, that's the limit right there.

MEMBER PALMTAG: You have to look for coolability?

MR. CORSON: Yes, so the concern

coolability, that's a big thing that the guide talks about.

It does mention there's other issues potentially related to environmental qualification or dose that are dealt with it to some extent in Reg Guide 1.183.

But it's -- I'll be honest, it's less clear what would need to be done to demonstrate, you know, to come up with an acceptable limit.

That's, you know, we're far more comfortable with zero. Beyond that, we're not as comfortable.

And so, we didn't have as clear guidance and we're, frankly, struggling with what would be acceptable.

And so, that's where --

MEMBER MARTIN: He's making my point.

MR. CORSON: Yes, it's --

CHAIR BALLINGER: You're sending a very clear message.

MEMBER PETTI: Yes, you opened the door and that was a mistake.

MR. CORSON: Yes, it's --

MEMBER PALMTAG: I'm sorry, this is a big picture, I realize this.

So, what's the point of this? You calculate the dispersal, you calculate how much --

MR. CORSON: Yes, ideally --

MEMBER PALMTAG: So, how is it not allowed --

MR. CORSON: Again, ideally, you calculate zero. If you don't calculate zero, then I guess, you know, per the guidance, it is allowed.

But what is acceptable and how you demonstrate what's acceptable, that's the part that's not as clear in the guidance. It's not -- to be honest, we haven't come to strong positions on this which is why I think the guidance is a little unclear.

Some of this is going to have to be hashed out in the final version of the rule, obviously.

Some of it is going to be hashed out during our interactions with the vendors and with industry as well.

MEMBER HARRINGTON: With very low regulatory uncertainty -- or very high regulatory uncertainty.

MR. CORSON: Yes, yes, that's true.

If you disperse anything, it's higher regulatory uncertainty.

MEMBER PALMTAG: So, the guide is how to

calculate it, but there's really -- the limits are still unknown right now?

MR. CORSON: Yes, the limits are still unknown.

Some of the considerations are known, the coolability thing, certain methods that you could use to come up with acceptable criteria.

But there are unknown unknowns, you know, other things that we might not be considering that it's hard to say that these are the only things you need to consider. And this is the only limit.

So, it's a tough problem, obviously, to come up with, all the more so in a few months.

I think John Lehning has some comments as well from the more regulatory side of things.

MR. LEHNING: Yes, John Lehning from the staff.

And I think James did a great job of answering that.

But I would say, even after we know more, because of the plant, each plant may have a different configuration as well.

And so, we could never be in a position where we would be able to say, it's always going to be a 100 kilograms and that goes for every single plant regardless of the size of the core and the design of the vessel and how much of it may in a certain plant get carried out into the containment or where it might go in the containment.

Because I mean, again, it could be some equipment that's being relied on that, if it spills out and fuel may pile on top of a certain piece of equipment.

And so, that's why I think that this guidance -- we will know more and we are working to do research.

And there are some things where, like the recriticality, where we may be able to sort of dispense with this for a large amount of fleet.

And we'll do that where we can. But it's very likely, in my opinion, that on some of these matters, we will still be in a case where, and I think this is what the regulatory guide is trying to do, where it spells out, here are the performance based objectives that need to be satisfied.

And then, it's up to the licensees to demonstrate what, in their plant or the vendors to demonstrate for their type of fuel, whether they satisfy those performance based objectives or not.

And so, I think that's where this is

likely on some of these to maintain, even in the area after we get done working on it further.

MR. CORSON: And I'll also add, you know, philosophically this is a concern for above the transition break size where we're going to a little more realism in the analysis because we think this is a very low likelihood event, So that's another consideration and another struggle.

How strict should we be for something that we're expecting to be unlikely. Something that based on our analysis, our inspections, everything else, show is unlikely to occur.

I think that's where we're coming from where we say, you know, some dispersal philosophically is acceptable because this is such a low likelihood event. And if it's a small enough amount that should be fine. But then answering the question of how much is enough, that's where we haven't really come up with a good answer beyond just the philosophical, a few grams or a few kilograms is okay.

MEMBER PETTI: Remember, this is a classic case of flexibility versus regulatory certainty. And there's a push in one direction, a push in another direction. It's not always the same for every case. But my personal opinion is, this is a slippery slope in a difficult area where regulatory certainty may be more important than flexibility.

MR. CORSON: Yes, it's, you know, we've struggled with this a lot. You know, not just, we've been working on the rule for seven months, whatever it is. It goes way beyond that.

I mean, we've been focusing on this since, really 2006 since we saw those Halden test results.

MEMBER PETTI: Right.

MR. CORSON: We struggled --

MEMBER PETTI: And if you remember our letter on the RIL --

MR. CORSON: Yes.

MEMBER PETTI: -- a lot of us had concerns that there was enough atypicality in the existing database that until you did something more typical it could look very, very different. And you're trying to make, you know, regulatory decisions on this, it's just, it's fraught with --

MR. CORSON: Yes. I think, you know, to that point the challenge is, how do you design a prototypical test.

You're almost a hundred percent going to be limited to small segment of the rod. We're not going to build a facility just to create a LOCA and melt it down, so.

MEMBER PETTI: It would be years ago.

MR. CORSON: Well LOFT, yes. Yes.

MEMBER PETTI: No, no, no. I mean, also there was tests at PNNL because of full length rods.

MR. CORSON: Yes.

MEMBER PETTI: It's the same exact issue, but in the days there was a lot more money back then.

MR. CORSON: Yes. So I think, you know, we're going to go back to those days where we're building these gigantic multi-million dollar, probably billion dollar to be honest these days, facilities where we're going to do these tests. So there is always going to be, you're always going to have smaller segments.

How do you heat that internally, it's really hard to do that. TREAT, you can sort of mimic some of this stuff, but TREAT has its own limitations.

Jules Horowitz Reactor, they have some ideas for how they would do tests in that reactor, if they ever get it up and running.

MEMBER PETTI: Finish it.

MR. CORSON: If they ever finish it. You know, the point is well taken from the ACRS. We've heard similar thoughts from Industry that the tests aren't necessarily prototypical.

But they're what we have in coming up with a better test. A more prototypical test is not so straight forward. So we're left with making regulatory decisions on imperfect information.

MEMBER ROBERTS: Along with tests as far as how you do analysis. When you mentioned 400 kilograms gets out, where is that written down, is that --

MR. CORSON: Yes, so we --

MEMBER ROBERTS: Is that published?

MR. CORSON: Yes. We had, we did this analysis, the Staff did this analysis. We presented it at the NURETH-20 conference in Washington --

(Simultaneous Speaking.)

MR. CORSON: Yes. In Washington, D.C. last year. We were then invited to submit it to Nuclear Engineering and Design. So there is some minor changes to the, what was published for Nuclear Engineering and Design.

MEMBER ROBERTS: Does that appear in an internal document?

I mean, the thing about journal and conference papers, you know --

MR. CORSON: Yes.

MEMBER ROBERTS: -- you can only fit so much in the 15-page limit or whatever the conference does.

MR. CORSON: So no, we don't have more internal, we don't have a longer internal report. I will say we had grand ambitions to write a NUREG or something on that and then we got tasked with doing a rulemaking and --

(Laughter.)

MR. CORSON: -- and coming up with a PIRT to help support the rulemaking, so. Like a lot of other things you do all this great work and it's wonderful and then you don't have time to document it. And so, we at least got the conference paper out of it. And the NED article out of it.

I will say, Industry did very similar calculations and published a report. I think it was Oak Ridge and INL with the same core design, similar sort of methodology.

They did publish a report which might have some more details than what's in our paper. At least on the core design, which is the same. The methodologies are a little bit different but they are similar.

MEMBER ROBERTS: Okay, great.

MR. CORSON: And they came to similar conclusions. I think they had a little bit less disperse mass. Maybe like two or 300 kilograms. But it comes down to what assumptions you make in the analysis.

For our analysis I'll say we, at that time we did credit the spacer grids with limiting dispersal, so it certainly would be less than that amount because if you would take the spacer grids into consideration.

So yes, that, and that was just, like I said, I can't say this enough, that core design is not very realistic so take those, take that to mind when you think of those numbers. But that's what we have.

CHAIR BALLINGER: I think that Craig was actually being kind. If you don't get dispersal no problem. If you do get dispersal you have no regulatory certainty.

MR. CORSON: Yes.

CHAIR BALLINGER: And that's really a very bad position to be in.

MR. CORSON: Yes. I mean, I guess the alternative would have been to write the rule in a way that would prevent any dispersal whatsoever. That was not the decision that was made.

And so, to come up with a guide that provides regulatory certainty in seven months when we haven't had time to think of all the various things that go into it, to do all the calculations ourselves, it's not easy, not ideal. So we're left with very imperfect guidance for what happens if you use fresh fuel, which we acknowledge.

CHAIR BALLINGER: So that begs the question, given the fact that we've got this potentially very serious issue that could be dealt with, with a little bit more time, we have this restriction on the rule having to be issued by X.

We're going to, we're going to issue a rule which no matter what we do in that amount of time will have something along the lines of what you're saying in there. Is that the right thing to do?

MR. CORSON: Well --

CHAIR BALLINGER: I mean, it's not your call, I'm sure.

MR. CORSON: Yes.

CHAIR BALLINGER: But is that the right thing to do?

MR. CORSON: So it's really tricky. There is different considerations here. I think ideally I would say no that's not necessarily the thing to do.

But this rule is much bigger than just fuel dispersal there is a lot of other stuff in this rule that if we would delay the entire rule just to deal with fuel dispersal it would have other impacts. And Industry has told us a lot, that this rule, and having it done on schedule, is very important to them. So we are somewhat constrained in what we can do.

I will say we, you know, requested an extension to our schedule. A lot of it was to deal with more of the stuff that Dave and Rob talked about yesterday, but was to help us as well. And, you know, we got a little bit if a longer schedule but not really.

It was a, an extra three months I think is what it ended up being. So we asked. We were kind of accepted, kind of denied. And so, yes.

DR. BLEY: Dennis Bley again. So the conundrum that's been hanging us up here at the end, have you heard anything from Industry about their feelings?

I mean, it seems to me they're in the spot that you got the rest of the rule and here you're either going none or you really take a gamble. What are the feelings that you've heard back from stakeholders?

MR. CORSON: So the unfortunate thing is that we did not have a time to have public meetings to discuss this specific issue. So I think we'll hear more from them during the proposed rule public comment period.

DR. BLEY: Okay. I'll bet.

(Laughter.)

DR. BLEY: Though I don't know what they'll have to say.

MEMBER HALNON: This is Greg Halnon.

(Simultaneous Speaking.)

MEMBER HALNON: This is Greg. Just a quick question. Do you guys, have you -- or would you consider maybe a graded approach to the level of analysis required or level of justification required?

I mean, I understand that you either have dispersal or not, but if you have some, very little, it may not require as much analysis and justification as if you have a lot. And the reason I say this because it just seems like there is a lot of margin and a lot of places that were, say is there, but we're not really crediting the rationale for why it's okay.

And maybe we need to go through the public comment, think about, you know, whether it should be a window of, in this window of kilograms, or whatever you want to call it, whatever parameter. You don't have to go off and do tests or something like that, but you have to justify it. But it just seems like there could be an opportunity for a graded approach here.

MR. CORSON: Yes. I think, you know, in theory yes, I agree, it would be possible to do this. As John Lehning had said earlier, it's hard to say what those thresholds should be though to even do less analysis or more analysis. And that, this is where we're struggling. It's also probably plant specific, as John had alluded it. It's, yes.

Again, in theory I agree with you. In practice, coming up with those limits, not so straightforward.

MEMBER HALNON: Well everybody is going to be in a learning situation and there going, you know, the applicants are going to find different methods, different ways, different models to be able to justify the amount. And it just seems there could be credit provided for very small issues, but maybe a good learning process.

I know you have to get engaged with the vendors and the Industry, and I think that's going to be pretty eye opening so we'll see where it goes.

Thanks.

MR. CORSON: Yes, we're certainly looking forward to the public comment period I would say.

MEMBER PALMTAG: This is Scott. I just want to follow-up on that. I kind of agree with Greg. And at first, I think you've well thought out your processes.

And to me it looks like at least you have enough data to get started. I mean, you always want more data but it seems like you have a pretty good limit on the data.

But once you start opening the door to these big fluid beds it gets really complicate. If you could maybe allow a one percent or five percent failure maximum I think that's a lot easier to digest.

You can learn down the road if someone wanted to go more they could do the analysis. But one or five percent failure seems like a much better target than these huge failures.

MR. CORSON: I just --

(Simultaneous Speaking.)

MR. CORSON: No, I agree. Again, I do back to what is, coming up with those limits, what's the basis for it, why is one percent okay and why shouldn't it be less than that, why should it be more than that.

You know, these are the things we're struggling with and it's hard to come up with these parameters. And again, it might be somewhat plant specific.

DR. BLEY: Well this is Dennis Bley again. It seems to me you don't need to come up with why. And three percent is not okay, you really just need, with some level to say, you know, one percent or whatever it is, some small amount is okay to, at least by engineering judgment can create more judge. It won't create and make a problem.

And the other side of that is to the places you're never, well at least not for a long time, is going to get an answer.

MR. RAYNAUD: This is Patrick Raynaud of the Staff. I just want to point out that the existing limits on PCT and ECR allow for fuel rod burst. And so, you know, the discussion of allowing no dispersal, even under current burn-ups, you can't say that it's absolutely zero. Even though you have low burn-up fuel you already have a greater than zero amount of dispersal. And so I think that saying no dispersal at all is not really realistic.

When it comes to fuel rod ruptures, you

know, I think even existing LOCA now that's used within the current criteria would probably result in more than one, five, ten percent of the rods bursting, depending on the transients and other things. And, you know, we can't really limit that to less than one percent or whatever unless we prevent all rod bursts. And that is much more restrictive than the existing PCT and ECR criteria that we have.

So I think some of these questions have to be more narrowly qualified. Maybe, you know, what dispersal did we not want to allow? Maybe it's just high burn-up finely fragmented fuel that we don't want to allow. I don't know.

If there is some bounds that probably need to be better defined if we go down the avenue if they're saying something like, no dispersal. Well, what kind of dispersal are we talking about specifically because I don't think even within our current regulations we can guarantee no dispersal.

MEMBER PETTI: Yes, but I'm a little confused. If when you're above the transition break size you are now best-estimate. And so, I agree with you if the statement was, I had to use all the conservative stuff. But now I, there is a lot of margin there that you could impose a more severe criteria than what was under the old rule set because you're doing a best estimate, so, you know. You got to look at it, the rule is set in the criteria together, not just the criteria in and of itself.

MR. CORSON: Yes, I'm not sure exactly what the final answer that a vendor would come up with using best estimate would be. I don't know what their final numbers on rod burst would be. I don't know if they can get to zero.

MEMBER PETTI: Well we've seen stuff that suggests that --

DR. BLEY: Yes.

MEMBER PETTI: -- A, they can, and B, the margin, relative to the peak clad temperature is large. Not 10 degrees, not 100 degrees, it is very large. Which is why there's all this discussion about power uprate because now I have margin and I can use it. Why wouldn't you.

MR. CORSON: Yes, I mean I, you know, I talked a little bit about our older analyses that use more of the realistic assumptions. Maybe not quite where Industry would go but, yes, we saw that for some plants that we just didn't get any ballooning in bursts let alone for low burn-up stuff. So it's certainly possible.

The way we wrote the rule did leave the door open for some dispersal. I think, you know, whether that's appropriate or not I think we'll get more feedback as we go along from the public comments and so on.

Again, I'd say our preferred path is no dispersal period. And, you know, the guide, maybe we don't make that clear enough in the guide, but certainly there is a lot more work to do and a lot less certainty, or no certainty perhaps for if you did disperse anything. But yes, that's where we're at.

We do, you know, we got feedback from Industry that no dispersal should be feasible during the regulatory basis public comment. And so as far as we know this might not necessarily be a big issue. But we wanted to have some flexibility because we don't know what the future holds. Doing rulemaking takes a long time, it's a big effort.

MR. LEHNING: And John Lehning from the Staff. Just to add to that too. I think this regulatory guide isn't necessarily the only item that would sort of inform how licenses evaluation plant go in the same manner that there are evaluation models that vendors propose in topical reports for the loss of coolant accident.

As it stands right now we have envisioned that there would be development of additional methods that Industry might come up with, that Staff would interact with them on. And then that might help better define, in more detail, how each vendor or licensee might do these calculations.

So I think we don't envision this as the only source of guidance, but I think some of the points, we understand like this idea of having some kind of limit, for now. I think we didn't do that, like James said, because it would just be perspective in one certain value or another. But we do see the data at this time that would support development of methods being along those lines of what could be justified as probably toward the lower end of the spectrum.

But I think the overall rulemaking approach here is predicated upon this takes a long time and we wanted to establish this pathway. And for right now the data limitations may require or may cause the vendors and licensees to be somewhat restricted or governed by some lower end.

But as they develop more and more data we'll have setup this rule approach and then the guidance modified. And then these methods can be developed by vendors that the staff can interact with them on, and then we can work through exactly where do we end up in terms of what's acceptable for these different plant configurations. So that's sort of just a little bit more philosophy of where this is coming from.

CHAIR BALLINGER: Questions? More questions? Okay, it is 11:40 and we're sort of on schedule. We're early actually. But we have lunch that's scheduled from 12:00 to 1:00. So let's recess until 1:00 p.m.

(Whereupon, the above-entitled matter went off the record at 11:40 a.m. and resumed at 1:00 p.m.)

CHAIR BALLINGER: Okay, we're back in session. Elijah, who is going to do the --

MR. DICKSON: I'll do the presenting, the talking.

CHAIR BALLINGER: Okay. Does Staff want --

MR. DICKSON: Do we have a Staff Member running through the slides?

PARTICIPANT: Elijah --

MR. DICKSON: Thanks. Yes. So want to go ahead and get started?

CHAIR BALLINGER: Yes.

MR. DICKSON: All right, great. My name is Elijah Dickson. I'm a senior reliability and risk analyst in the division of risk assessment. And this is my co-worker Dave Garmon, health physicist. We'll be presenting today on the increased enrichment rulemaking with respect to the general design criteria of GDC-19 in 10 CFR 50.67.

With that, we'll go ahead and go onto Slide 2 please. So the topics today, we'll be discussing the rulemaking driver and goal just very briefly. We'll give a little bit of background in history into the control room design criteria itself. We'll discuss some of the regulatory issues with respect to what we're seeing now with current enrichment and burn-ups and what we expect with higher burn-ups and increased enrichments. Then we'll discuss the rulemaking proposal language itself. We'll walk you through our proposed changes to the rule and our rationales for the proposed changes. And then lastly we'll discuss our rulemaking approach so you can understand our decision making process.

So with that let's go ahead and go on to Slide 3. The legislative driver for proposing an amendment to the control room design criteria is the Nuclear Energy Innovation and Modernization Act signed into law back in January 14th, 2019. Purpose of this proposed rule in response to the nuclear power industry's interest in accident tolerant fuel, or ATF, in conditional fuel designs with higher enrichment as well as interests in economically beneficial operational targets to meet the standard cycle times as well as perhaps power uprates as well.

So what the Staff has done is evaluated several areas of the regulatory framework that would, could be impacted with increased enrichments and higher burn-ups and assess whether or not adjustments can be made in order to support efficient reviews of these licensing actions.

With that, let's go on to Slide 4 please. A little bit of background in history of general design criteria for the control room. Both GDC-19 and 10 CFR 50.67 provides specific dose base criteria in a 5 rem total effective dosage equivalent, or TEDE, for demonstrating compliance with control room acceptability. These rules represent their own distinct layer of defense-in-depth that assumes a major accident results in a substantial meltdown of the reactor core with a subsequent release of appreciable quantities of fission products to the containment.

These are your classic performance-based regulations which require the licensee or applicant to provide a control room accountability design using traditional deterministic radiological consequence analysis methods to judge its acceptability. The performance-based aspects of this rule are typically covered in regulatory guides.

Specifically the topic tomorrow will be Reg Guide 1.183. We have a draft guide on that. It's referred to as BG-1425.

It's important to note that both the general design criteria under GDC-19 and 10 CFR 50.67 are design criteria. And they should not be construed as operational limits.

The design criteria itself do not represent actual occupational exposures received during normal operations or emergency conditions which are primarily controlled, are at 10 CFR Part 20, the standards for protection against radiation. Also, the consequence analyses themselves are more than just an initial design analysis or an academic exercise. They do play an important role in the day-to-day operations within the nuclear fleet. Such as verifying other regulatory requirements, guiding maintenance activities, which do incur actual occupational exposures, as well as serve as a guideline in performing a 50.59 analyses as well.

Let's go on to Slide 5.

MEMBER ROBERTS: Wait, Elijah. Tom Roberts. When you looked at the history did you point out why the word accident instead of design basis accident is used in GDC-19?

MR. DICKSON: No, I couldn't find that.

MEMBER ROBERTS: And one of your public comments questioned that in whether the, you know, the guideline or at least part of, you know, related to severe accident doses. I guess what your conclusion is, is practice has been, it's a design basis accident dose not a severe accident dose.

MR. DICKSON: Yes.

MEMBER ROBERTS: I was wondering if you found anything that would, you know, prove that?

MR. DICKSON: Yes. These are design basis accidents. We do use what's referred to as a severe accident source term. That's something that we'll be talking about tomorrow in some of the history behind the in-containment source term.

There are a few discussions in regards to the purpose of this criterion in the statements of consideration for 50.67 that was published in 1999.

We'll discuss that.

MEMBER ROBERTS: Okay, thank you. It's a severe accident released into containment but it doesn't model the several accident phenomena could in fact continue, right?

MR. DICKSON: That's correct. That's right. Yes.

MEMBER ROBERTS: So it's arguably either very conservative or somewhat not conservative depending on what accident scenario you're looking to assess against.

MR. DICKSON: Right. We call that a representative source term for design basis purposes.

MEMBER ROBERTS: Right. Okay, thank you.

MR. DICKSON: Yes. Okay, so we're on Slide 5. All right. So in moving forward with increased enrichment we, typically what you do is you assess the acceptability of our current regulations. Right now the control room design criteria is one of three cited, three dose base criteria that we use in these analyses.

The first being the exclusionary boundary, or EAB. We have the low population zone and the control room. The control room right now is the most limiting figure merit or fact, between the three.

So during the development of the original GDC-19 in the late 1960s and early 1970s they didn't foresee, or staff didn't foresee how licensees are currently operating their facilities and managing their fuel. The history of fuel utilization with the fleet has seen a gradual progression towards higher fuel discharge burn-ups and higher enrichments.

There has been enough margin in the facility's design basis to effectively accommodate the control design criteria up to 120 percent, for instance, of their originally licensed steady state thermal power. So during the power uprates about 15 years ago we were able to accommodate this, this criterion.

The NRC recognizes that there are challenges that the licensees face with obtaining margin within their licensing basis for the purposes of operational flexibility and the small amount of margin to the control design criteria itself. In general, not even specific just to the control room design criteria, it's an unjustifiably low design criteria used in regulations can unnecessarily burden licensees seeking increased enrichments by at times requiring extensive analyses to preserve margin.

Often, just from licensing experience,

it's typical if licensees need additional margin in their consequence analyses that will sharpen our pencils before re-analyses. And sometimes they can be extensive analyses. At times they can deviate from regulatory guidance, the staff then needs to double check the models, reach out to our staff members in the Office of Research, make sure that they're all right.

And so what we're looking to do here is make sure everybody stays on the same page in regards to looking at these increased enrichment technologies and utilizing the regulatory guidance that we got.

MEMBER PETTI: Elijah?

MR. DICKSON: Yes.

MEMBER PETTI: Just a question. I understand this idea that the margin to the design limits are getting tighter.

MR. DICKSON: Right.

MEMBER PETTI: But we're doing two things at the same time. One is, you're talking about losing those, but the source term, that we'll worry about tomorrow --

MR. DICKSON: Right.

MEMBER PETTI: -- is going tomorrow. Is it about 50/50? Is the source term going down providing more greater margin? How is that in there?

MR. DICKSON: I think we'll answer that question I think between today and tomorrow. At the very end of this slide I'll sort of talk about our three-prong approach to this rulemaking efforts.

I think we can go ahead and just cut to the chase. That we did, we're looking at this in three different ways.

One, making improvements to the source term itself, that in containment source term, expanding its applicability to reach licensees operational targets to go to these longer cycle times. So that means we work with the Office of Research in developing this updated Sandia 2023 source term and these pathway specific source term models, right.

So then the second prong of this approach is to update the transport models themselves. Several of these transport models can date back to the 1960s in these modeling practices. And so we had discussions last month in regards to developing these pathway specific transport models that can credit safety related features to mitigate the source term that weren't technically credited.

And so, and that's one area in this, in the area of consequence analysis that really hasn't had a lot of attention just updating these transport models. We spent a lot of time and attention on the in-containment source term itself.

And then the last part of this three-prong approach is to effectively update the control room design criteria itself. And that's the discussion today. Presenting to you are multiple lines of reasoning as to why we believe it's okay to propose an amendment to amend the control room design criteria by essentially by a factor of two. Okay.

MEMBER PETTI: Thanks.

MEMBER ROBERTS: Elijah, I'll get to this question eventually, but when I read this whole section what concerned me was that the design criteria is using a somewhat artificial source term. And in reality you'd like to have this habitable condition that's practical in the control room for virtually any, you know, plausible scenario which would include certain accident scenarios like they say in Fukushima.

And so, when you get through this, how much of this is pencil sharpening and how much of this is actually allowing a leakier control room or, you know, a relatively higher dose in the control room for accident scenarios that aren't part of the way that you're treating the regulation?

MR. DICKSON: Sure thing. So it's important to remember that this is a design criteria itself. That it is one part of the overall regulatory framework that demonstrates adequate protection.

We went through, we spent a lot of time and attention in the federal register notice, as well as some of these other white papers that we presented, that discussed the radiation protection framework itself, as well as the emergency response plannings. All of that together provides adequate protection to the control room operators and other folks that are, you know, performing emission doses during actual emergencies.

We have evidence that this does work quite well. Fukushima being a good example of a very, very beyond design basis accident occurring in which the design basis of the facility was completely moot. Especially the control room itself in an extended loss of offsite power accident where they were able to actively respond to the emergency at hand and ration out the mission doses that were received. And they did a very good job with that.

Both UNSCEAR and the IAEA have published reports on that operational experience and how they dealt with maintaining doses of ALARA, essentially.

And they did a very god job trying to work with upwards to several thousand people onsite for that accident.

MEMBER ROBERTS: They we were also very hindered by leakage into the control room.

MR. DICKSON: What's that?

MEMBER ROBERTS: They're also very hindered by leakage into the control room.

MR. DICKSON: Yes.

MEMBER ROBERTS: There were reports to make masks, you know, 24/7. And had to take some heroic actions for ALARA.

MR. DICKSON: Right.

MEMBER ROBERTS: I mean, they still got some pretty high doses --

MR. DICKSON: This is important to remember, that was design basis. I mean, that's far beyond design basis, not design basis. At some point you do need to say, what is like reasonable for design basis purposes. And that is what has been presented to, you know, with us over several, I think two years now at this point, with that in-containment source term. That the magnitude of that source term is good enough for the purposes of designing the control room.

MEMBER ROBERTS: Right. And if you were

dealing with things like power uprates and details in the transport model and that kind of thing I get that, but if the upshot of a change like this is to allow a leakier control room because that's now permitted per the requirements, you know, I would think you would also need to look at what that might do in terms of the challenge of managing the spurious unit. Because as you point out, the emergency planning and the framework does allow for lots of things you wouldn't do normally. And it also, you know, support, have people onsite to --

MR. DICKSON: Right.

MEMBER ROBERTS: -- event, and it will do whatever you need to, to keep the people safe within the confines of what the mission is.

Again, starting from a condition with a less tight control room --

MR. DICKSON: Right.

MEMBER ROBERTS: -- just adds to that challenge. And then that's the part I didn't really see --

MR. DICKSON: Got it.

MEMBER ROBERTS: -- discussed in the white paper. That the upshot is, again, to better model transport, you're great, but the upshot is to have a leakier control room which would have, in practice, you know, a higher control --

MR. DICKSON: I'm not sure if that's the --

(Simultaneous Speaking.)

MEMBER ROBERTS: -- that seems like, you know, we would want to avoid.

MR. DICKSON: I don't think, I appreciate that. And I think that's some things we avoid too. Designing a leaky control I don't think is the intent of these analyses. Is one part of them.

The input parameters of these types of analyses or not just assessing the leak tightness of the control room, they're also assessing the leak tighteners of the valves within the facility, the leak tightness of the containment itself. They do have other uses as well, not just control room.

MEMBER ROBERTS: Yes, I agree with all that.

MR. DICKSON: Yes.

MEMBER ROBERTS: I mean, that's all, you know, necessary part of the regulatory instruction --

MR. DICKSON: Right.

MEMBER ROBERTS: -- to make sure that the containment meets its design requirements because there is also scenarios where we don't, won't --

MR. DICKSON: Right.

MEMBER ROBERTS: -- because of the nature of the accident.

MR. DICKSON: That's right.

MEMBER ROBERTS: And what you have left is the protection of the operators of the control room.

MR. DICKSON: Right.

MEMBER ROBERTS: And again, if the upshot of the exchanges were to make that environment worse because it's allowed by the regulation then that's something I think we ought --

MR. DICKSON: Yes, I'll push back with those types of words of saying that we're making the environment worse. I do not believe we're making the environment worse with providing some relaxation with the control room design criteria.

It's important, it's really important to remember that these are not occupational dose limits. We have that controlled under Part 20. Absolutely.

MEMBER ROBERTS: Okay, thank you.

MR. DICKSON: Thank you. Okay. So part of the discussion we'll be having tomorrow as well is the regulatory guide itself and the in-containment source term. And we'll be discussing what both the

NRC and the Industry have been doing now for about 40 years at this point, and that is transitioning the originally licensed TID-14844 source term that was developed back in 1962 with an alternative source term that was published in NUREG-1465 in 1995.

We have been transitioning the fleet. At this point we've done a hundred percent of the fleet, I think. There might be one or two reactors that have not transitioned to the alternative source term.

This re-base laying action, these are very large complex multidisciplinary type reviews. The license amendments that come in can often be several thousand pages long for licensees.

The engineering analyses and the consequences analyses can take long lead times to develop and prepare. And they can take contractor support as well.

From the staff's perspective. From the staff's perspective we have, I just lost my notes. We spent several decades reviewing these licensing amendments under 50.67 to adopt the alternative source term. We have a tremendous amount of licensing experience in this area.

And we would like to preserve all that kind of great work. And we're actually proposing some great improvements we think, not only to the rule itself but to the guide and to help us move forward in this particular area of the regulatory framework. And as I mentioned, at the very end of the slide I was going to talk about our three-prong approach and the control room design criteria itself was one of those particular areas that we're going to look at.

So let's go ahead and go on to Slide 6. So the regulatory issues. When considering rulemaking we need to research the original bases for the rule to understand the Commission's reasoning for developing it in the first place. And once we understand the bases for developing it we can then make informed decisions as to whether or not he rule meets today's needs, right? This particular environment.

From there we can assess if some relaxation could be warranted and still provide a reasonable assurance of adequate protection. With the exception of, Member Roberts, with the exception of the 1999 rule, the original bases for the control room design criteria itself is not readily available.

We were able to find early drafts of the design criterias published in the late '60s, as well as some meeting minutes, some correspondence between staff. But the numerical selection for that, the basis for that numerical selection is, wasn't directly addressed.

Effectively what they did is they referenced at that point, or that point in time, the part wouldn't, Part 20 radiation protection occupational exposure limits at that point. So it's an easy thing to reference. It's there, why not reference it as a design criteria.

Now in the preamble to 10 CFR 50.67, which was published back in 1999, there is a little bit of information regarding the Commission's rational for the original establishment of the GDC-19. And the first sentence says it all. But then it discusses what the purpose is of these criteria, right?

And I'll go ahead and read it verbatim. That the criteria of GDC-19 were based primarily on the occupational exposure limit. That the use of the 5 rem TEDE as a control room design criteria did not imply that is the value that would be acceptable during exposures during emergency conditions or that other radiation protection standards of Part 20, including the individual organ dose limit might not apply. The criterion was provided only to assess the acceptability of the design provisions of protecting control room operators under postulated DBA, or design basis accident conditions.

The DBA conditions assumed in these analyses, although credible, generally did not represent actual accident sequences but were specified as conservative surrogates for creating boundary conditions for assessing the acceptability of engineer safety features.

So with that, let's go ahead and go onto Slide 7. We've established now what the purpose of the control room design criteria is. And we'll do a little bit of a refresher in regards to the regulatory basis document and the different alternatives that we assessed in ultimately going down the path of what we decide to do following the public comments that we received.

So the first alternative was to maintain the current regulatory framework. The NRC would continue to revise the existing regulatory guidance on an ad hoc basis as the research results and resources become available.

Licensees would use regulatory guidance or very likely develop their own regulatory guidance. These are some of the things that we were hearing about maybe four or five years ago that they were having difficulties meeting the control room design criteria as specified in the consequences analysis and the regulatory guide. It is likely that the Staff would need to review these on a case-by-case basis as they went through the licensing actions.

Alternative 2. The NRC would pursue rulemaking to amend the control room design criteria with stakeholder input and identify and assess a range of acceptable values based on sound regulatory and scientific insights. This would also leverage severe accidents research to develop an updated AST to meet the operational targets the fleet is looking at, as well as develop updated transport models and read baseline several other operational and human health assumptions and guidance.

Alternative 3. That is our, effectively our research heavy, research heavy approach where the regulation would remain unchanged. This would initiate new staff initiatives to develop other mathematical methods, computational and statistical analyses and perform consequence analyses.

This would have taken we believe several years. And the results of that work would still need to be disseminated out to stakeholders, develop the guidance.

Come to ACRS to present on these new

methods. And ultimately we felt the presented, if they were still acceptable at that point, in a Revision 3 a regulatory guide 1.183.

Okay. So let's go ahead and go on to Slide 8 please. Public comments. In the regulatory basis document the NRC sought comment on each of the proposals. We Recommended Alternative 2 in that regulatory bases comment.

And we also asked two questions. The first question sought input as to whether the numerical selection of the control room design criteria would be better aligned with regulations designed to limit occupational exposures during emergency conditions or regulations designed to limit occupational exposures during normal operations.

The second question sought to input as to whether a graded risk-informed method for demonstrating compliance with a range of acceptable control room design criteria values instead of the single value would provide the necessary flexibilities for current and future nuclear technologies.

Overall, public comments were supportive of a Staff's recommendation to amend the control room design criteria under Alternative 2. They generally suggested a value of 25 be applied to the amended regulation.

Nearly all commenters included suggestions to develop a graded risk-informed approach to the control room design criteria. Several commenters commented, relied on PRA technology and methods and contemporary understandings of facility risk profiles to justify a higher numerical value for a low probability high consequence event.

As Member Roberts brought up, there was one submitter who was not supportive of Alternative 2 or Alternative 3 and preferred to go down the Alternative 1 pathway. And based off of this commenters comments, we felt that it was very important to go very heavy in radiological risk communication.

Explaining what the regulatory framework is for radiation protection, as well as framework to response planning and then discuss, at length, what these numerical values actually mean in regards to health effects.

So you will see that reflected in the federal register notice, as well as in these other white papers just to, trying to provide some context as to what it is that we're trying to accomplish here in this rulemaking effort. And we also wanted to make it very clear that we are not changing the operational exposure limits of Part 20 or its underlying radiation protection philosophies that we've adopted through the ICRP.

So with that, we can go on to Slide 9. All right. So we down the alternative to regulatory basis item, I supposed. Within Alternative 2 we identified three options. Options 2A, 2B and 2C.

Options 2A and 2C are similar. They would simply amend the control room design criteria to either 10 rem or 25 rem. Option 2A was 10 rem, Option 2C would be 25 rem.

A value of 10 rem we felt wouldn't provide the additional flexibilities licensees may require and could lead to some possible exemptions if you wanted to go above 10 rem. But then the 25 rem would provide the maximum amount of margin that the Staff was really comfortable.

So we also had this option, 2B, which is a combination of the two. Where you can codify the numerical acceptance value by a factor of 2 from 5 rem to 10 rem, but then allow for a graded approach to reaching 25 rem based off of the facility specific risk information. And that's how we've gone down this path now at this point.

So let's go ahead and talk about the proposed rulemaking language in Slide 10. Oh, sorry. Yes, Slide 10.

High-level changes. We're proposing a modest increase of a factor of 2 from 5 rem to 10 rem. All nuclear power plants in the United States would benefit from a higher but still safe performance level from 5 to 10 rem when implementing advance pool technologies and maintaining their operational flexibilities.

If additional margin is needed beyond 10 rem they can use facility specific risk profiles or information that can be leverage to just by a higher numerical value.

We're also clarifying the purpose of a control room design criteria to distinguish it from the radiation protection and emergency preparedness framework with some proposed textual changes.

The proposed amended values will, ranging up to, between 10 and 25 rem would be consistent with aspects of 10 CFR Part 20 which allow for any frequent occupational exposures of the 10 rem. And those that are discussed in 10 CFR 50.47 emergency plans which provide provisions to follow the EPA PAG guidelines which go up to 25 rem depending on the severity of the accident.

Also, the upper range of proposed numerical value would be consistent with the use of 25 rem which is applied in other regulations such as the exclusionary boundary and low population zone of power reactor siting which is generally thought to be assessing public protection during emergencies.

Then lastly, GDC-1425 would adopt a method and develop a framework for a graded risk-informed and performance-based control room design criteria.

Now I'm going to walk you through some of the proposed language on Slide 11.

MEMBER MARTIN: Real quick.

MR. DICKSON: Yes.

MEMBER MARTIN: It's more of an academic question. In considering revising design criteria --

MR. DICKSON: Yes.

MEMBER MARTIN: -- do you know the EAB has a time as a clock?

MR. DICKSON: We're going to talk about that too. Yes. The --

MEMBER MARTIN: Okay.

(Simultaneous Speaking.)

MEMBER MARTIN: -- consider a clock?

MR. DICKSON: Oh, you mean like consider

a clock in changing the acceptance criteria itself for the control room?

MEMBER MARTIN: Yes.

MR. DICKSON: No --

MEMBER MARTIN: I mean, like --

(Simultaneous Speaking.)

MEMBER MARTIN: -- expect maybe a shift 12 hours or something like that.

MR. DICKSON: All right. There in the consequence analyses we do have what's known as occupancy factors. And there are occupancy factor assumptions that go into that.

And we're not talking about it directly tomorrow in the updates to the reg guide, but what we are doing is providing a provision in the guide that allow licensees to make their own occupancy factors based off of their actual, you know, emergency planning --

MEMBER MARTIN: Right.

MR. DICKSON: -- plans, right? But yes, we codified in these rules, the other acceptance values that we look at, the exclusionary boundary is 25 rem over two hour periods.

MEMBER MARTIN: Okay.

MR. DICKSON: It's a pretty high dose rate as compared to, what we have right now is 5 rem over a 30 day period. And low population zone 2 is a 30 day period as well. Yes, that's all I got on that.

MEMBER MARTIN: Okay. No, that --

MR. DICKSON: Yes.

(Simultaneous Speaking.)

MEMBER MARTIN: -- you've obviously thought about --

MR. DICKSON: Yes.

(Simultaneous Speaking.)

MEMBER MARTIN: -- question.

MEMBER ROBERTS: Elijah, if I get a chance to ask about the process for making these changes. We spent the last day and a half closely looking at FFRD. One of the things that was clear there is the approach is, there is a pretty detailed process for making changes enabled by the rule.

MR. DICKSON: Yes.

MEMBER ROBERTS: I didn't see any of that here. And in fact, there was a sentence that used to be in the Reg Guide 1.183 that seemed to be deleted that wasn't prior specific approval to make any design change that would be enabled by the AOC.

So is your thought that once you put this out there you wouldn't need to get to the justification to approve the design changes that were enabled by the higher control room dose or the different model?

MR. DICKSON: In rev, I'm not sure what specific sentence was deleted in Rev 1. Though in Rev 1 we had discussions about how licensees can make changes on their own through 50.59 as long as they meet the acceptance criteria. It's in an NEI document. It escapes me right now.

It's an NEI approved document, we're going through 50.59 changes, where they can make a change if their analysis of record is within ten percent of the acceptance criteria in the regulation itself. And if they're within that ten percent they can make the change on their own.

But we did retain, I'm almost certain in Rev 1, language that talks about making changes to the technical specifics. That if there is going to be a change made to the technical specification, even if the change meets that 50.59 criteria they still need to come in with the license amendment.

MEMBER ROBERTS: Yes, we can talk about it tomorrow, but the sentence that I highlighted that was in Rev 1, but is not in Rev 2, it says, licensees should evaluate proposed modifications that seek to downgrade, remove, require engineering safeguards equipment to confirm that the modification does not invalidate its functional state at the facility, PRAs does not adversely impact the facility severe accident management program.

So I couldn't find that sentence in Rev 2. It seems to be deleted. So we can talk about that tomorrow when we get --

MR. DICKSON: Yes, sure --

(Simultaneous Speaking.)

MEMBER ROBERTS: -- reg guide.

MR. DICKSON: Yes. Thank you.

MEMBER ROBERTS: That kind of that gave back my, you know, my question about the leak tightness in the control room. Somebody were to use this criterion to have a leakier control room, that that would appear to give you a hook to at least, you know, get another look at that.

MR. DICKSON: Yes.

MEMBER ROBERTS: But taking that hook out of the reg guide would then, I think that's something that would be within our scope to do.

MR. DICKSON: Okay.

MEMBER ROBERTS: So again, just to, you know, to try to make sure that the severe accident and worthiness of the control room isn't degraded by these rule changes.

MR. DICKSON: Understood. Yes. Thanks for giving me some homework tonight.

(Laughter.)

MEMBER ROBERTS: Okay. Yes, thank you.

MR. DICKSON: Yes. So let me walk you through, is there a question over there, Member Martin?

MEMBER MARTIN: No.

MR. DICKSON: No? Okay. All right. Let me walk you through the rule change. Or the proposed rule change.

And I'm going to discuss more rational for these proposed rule changes. So the original phrase of adequate radiation protection would be replaced with the necessary design fabrication construction testing and performance criteria for structure systems and components in order to safely keep, which is consistent with the purpose of the discussions in the opening paragraph of Appendix A, to 10 CFR Part 50 the general design criteria.

As explained in the opening paragraph, SSCs important to safety are those SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

However, over time the appraised adequate radiation protection has been completed with NRC's statutory authority of adequate protection. Adequate protection is achieved through the licensee's compliance with the NRC's comprehensive radiation framework of which the GDC is one part not just one particular design criteria.

Also, we are proposing the phrase personnel receiving to be replaced with the phrase calculated and the phrase access and be deleted. The performance of the control room design criteria does not assess actual personnel receiving radiation exposures nor does it assess personnel traveling to and from the site boundary to the control room. We believe that these will help address some of the licensing issues that we received over the last, I'd probably say 20-ish years in licensees adopting the alternative source term.

And lastly we are proposing a new paragraph, Paragraph 3, which would provide the additional operational flexibilities if needed be on 10 rem up to 25 rem. And execution of this paragraph would be provided in guidance. And we'll be discussing that tomorrow.

Okay. So now that we discussed the bases and the purpose of the rule --

MEMBER KIRCHNER: This is Walt Kirchner.

MR. DICKSON: Yes, Walt.

MEMBER KIRCHNER: Could I interrupt for a moment while this is up?

MR. DICKSON: Absolutely.

MEMBER KIRCHNER: Could you go over what was deleted and why? For example, access was deleted.

MR. DICKSON: Yes. So in, access can be implied in a couple different ways. Access can be just opening the control room door and getting into the control room. The ability to do that.

It could also be interpreted as access from the site boundary to the control room itself. And so, when you do the consequence analyses, what you can interpret the phrase access as having to have gone for the dose received to the dose receptor during that period of transit.

MEMBER KIRCHNER: Well as a former operator that's the consideration.

MR. DICKSON: That's right. We do have TMI action plan items in NUREG-0737 that talk about the use of mission doses. I mean, these are assessed in a EPR-20 type framework. Does that answer your question potentially?

MEMBER KIRCHNER: Well I think the original concept was that the actual operators would not receive a dose in excess of 5 rem TEDE. Well the TEDE wasn't there originally, but --

MR. DICKSON: Right.

MEMBER KIRCHNER: -- during, for the duration of the accident, so it was designed conservatively. You could say that the 5 rem was arbitrary but it was a conservative estimate. So we're relaxing this.

MR. DICKSON: I think another way you could look at it took is that, what about having design provisions for operators that go from the site boundary to the control room. Could here be shielding from the site boundary to the control room? I've been told at times that that's been considered.

MEMBER KIRCHNER: Well if you are going to consider, as you suggested, that there could be a rotation in the manning to manage the total exposure and so on, then that would be a valid consideration in laying out such a strategy for relieving operators during the course of the duration of the accident.

MR. DICKSON: Okay. And then these are

just proposals for language. And I do hear what you're saying. Did you want to talk about the phrase calculated as well?

MEMBER KIRCHNER: Well what's the subtly that you're trying to make here? It's actually, the concern is an actual operator receiving the dose. This is calculated so I don't have a major problem with that.

Obviously the design of the control room and the prediction of the operations is based on shielding and over calculations. Obviously leakage rates, et cetera.

MR. DICKSON: Right.

MEMBER KIRCHNER: So the whole exercise, for purposes of the existing GDC, and I guess for the new revised proposed language, would be a calculation. I don't have a major problem with that part.

MR. DICKSON: Yes, I think, I'm trying to think back to some of my research in the ASTs that we've received. As I mentioned, we've moved the entire fleet now at this point from the TID source term to the AST.

And I think I can probably think of three or maybe four facilities that account for this, it's called a, in the consequence analysis world it's called a travel dose. And these accounted for, like groundshine for instance on, in the contribution to the dose receptor within the control room.

Now often, if I recall, those values are usually pretty low. Most of the fleet. Vast majority of the fleet do not consider in this transit dose.

I can go ahead and move on to the next slide or we can keep discussing this if you like?

MEMBER KIRCHNER: No, go ahead.

MR. DICKSON: Okay, thank you.

MEMBER KIRCHNER: Thank you.

MR. DICKSON: Yes. All right, so now that we've gone over the bases for the rule and talked about the proposal to amend the control room design criteria and some of the proposed contextual language changes as well, I want to walk you through the Staff's decision making process and how we got here.

First what we did was we assessed the foundation of the Commission's policies and regulations concerning the radiation protection framework with this common understanding with how and if proposal to the control room design criteria is possible. From there we looked and research evidence-based justifications for proposed numerical values for the control room design criteria.

With this common understanding of those first two areas we can then proceed with developing reasonable regulatory relief through rulemaking. And so we'll go through each of these items for these next few slides.

Slide 13. All right, so let's talk about the foundational policies and regulations that Part 20 is based off of. And what we're trying to do here at this point is look for flexibility within the commission policies proposed to submit a control room design criteria value.

So Part 20 puts into practice recommendations from the International Commission on Radiological Protection, or ICRP, and its U.S. counterpart, the National Council on Radiation Protection & Measurements, the NCRP. Both these organizations are responsible for the advancements and state of knowledge and practice on all aspects of protection against ionizing radiation to provide recommendations and guidance for public benefit.

From this we went to the statements of consideration published in the federal registrar underneath, under 56 FR 23360 published in 1999, which discussed the purpose of 5 rem TEDE for the occupational exposure limit, which is to limit stochastic effects, or effectively to limit stochastic effects for cancer mortality for large populations and to prevent deterministic effects.

Three is flexibilities provided under planned special exposures as described by the ICRP, recognizing that under certain conditions exceeding the exposure limit may be needed either under normal or emergency conditions. The ICRP does permit up to 10 rem in addition to the annual limit, which can equal 15 rem in one year.

The NRC, when adopting the ICRP recommendations, it did intentionally reduce this value for planned special exposures from 10 rem to 5, which would equal 10 rem as an upper limit. And that's reflected in our current rules now.

The Commission deemed this to be a safe practice specifically because it would be infrequent. And for small groups, or particularly for individuals.

Part 20 also retained the lifetime total dose limit for planned special exposure to 25 rem, the same as ICRP recommendations, but it must be spread out over a period of time. No faster than 5 rem in any one year.

So our findings from looking at some flexibility within the Commission's policy is that there is flexibility for proposing an amendment to the control design criteria taking a risk-informed approach to this.

So with that, let's go ahead and go on to Slide 14. So this slide talks a little bit more, more highlights the radiation protection and emergency response frameworks highlighting that, it covers both normal and emergency responses conditions, as well as prepare licensees to respond to abnormal and emergency conditions to protect the public health and safety.

Nuclear power plant facilities are designed to ensure a readiness for a wide range of beyond design basis accident conditions. These regulations mandate comprehensive training, emergency response drills, exercises to equip plant operators and staff with the skills and knowledge necessary to effectively manage an event.

Our findings from this review found that there are a range of dose-based of occupational exposure limits, as well as design assigning criteria values that are also dose-based, up to about 25 rem within the regulation. And understanding the basis and purpose, and application of these rules, help inform the Staff in proposing this numerical change from 15 rem to 10 rem and ranging up to 25 rem for the control design criteria.

So with that we'll go and briefly talk about each of these rules over the next three slides. We're on Slide 15.

So at the time the GDCs were established in 1971 the Part 20 limit for the occupational radiation exposure was 3 rem whole body for calendar quarter provided that the lifetime dose was verified not to exceed in any specific limit, thus workers could receive up to 12 rem in any one year.

And you can see this when you look at the NUREG-0731 radiation exposure information reporting system report. You can look at exposures in the '70s and '80s and see very well that folks, on occasion, would receive actual application exposures above 5 rem, sometimes above 10 rem. It's just something that was common back then. Now that's not the case.

The fleet have doses like well below 3 millirem I think at this point on average. So the current annual limit for occupational exposures under 20.1201 is 5 rem per year, but technically a worker could receive an occupational radiation exposure of 10 rem over a 12 month period startling two calendar years.

And as I discussed with planned special

exposures, that individuals could receive an additional 5 rem above their occupational dose limit in one particular year for a maximum of 10 rem in one year.

Now let's go ahead and go on to Slide 16 talking about the emergency plans. Those are regulations in 50.47 do require licensees to establish the means for controlling radiological exposures during emergency.

The rules states that the means for controlling radiological exposures must include exposure guidelines consistent with the EPA's emergency worker and lifesaving activity protective action guides. The EPA exposure guidelines recommend that doses received under emergency conditions should be maintained ALARA and to the extent practical, below 5 rem.

The guidelines do provide flexibility however. Either in the severity of the event for protecting property values of 10 rem, and for lifesaving and protecting large populations up to 25.

All right, so let's go ahead and go on to Slide 17. The upper range to the proposed numerical value would be consistent with the Commission's use of the 25 rem, primarily regulations for power reactor citing. And you can find these under Part 100, as well as in two places in Part 50, 50.34 and 50.67. And I believe there is several areas in Part 52 as well.

As discussed in the preamble to the 50.34 rule that was published in 1996 in the federal register under 61 FR 65159, it provides a little bit of context in regards to the Commission's use of the 25 rem value stating that the Commission's use of 25 rem does not imply that it's considered to be an acceptable limit for emergency doses to the public under accident and conditions, but only that it represents a referenced value to the use for evaluating planned features and site characteristics intended to mitigate the radiological releases of accidents in order to provide assurance of low risk to the public under postulated accidents.

The Commission dates this, their rationale for continue to use this value based off of the extensive experience applying this criteria in recognition of the conservatism under the assumptions made in the consequence analyses. In particularly, a large fission product released in containment associated with a major core damage accident, maximum leak rates on the containment and conservative atmospheric dispersion modeling practices.

So with that, I've gone through some, to give context as to how the NRC utilizes the different dose days exposure limits and dose days citing criteria and design criteria. From there we can talk briefly about the Staff's research in regards to what are the evidence-based scientific recommendations out there.

So on Slide 18 the Staff reviewed several sources of the material to understand the current recommendations for national and international organizations responsible for making recommendations for radiation protection standards. The purpose of this review is to determine whether reexamining the scientific and technical bases for the numerical value itself would be warranted.

Findings, our findings from this review is published in a paper, a white paper, from Dr. Brock in the Office of Research. It's entitled, a Control Room Design Criteria and Radiological Health Effects. And you can find that at the ADAMS number at the bottom of this page.

And then general our findings from this review, as well as the continued work on the publication of this white paper is that there are a range of international and national organizations that are responsible for making radiation protection recommendations for both normal and emergency conditions up to 25 rem.

I'll be going to Slide 19 please. This slide simply presents some of the referenced documents that we reviewed. We looked at the EPA, PAG Manuals CORs, ICRP, IAEA, the NCRP as well as the Health Physicist Society. And I believe the white paper looked at a couple other sources of information too as well.

Okay. So let's go on to Slide 20. So as part of our risk communications with this proposed rule we included in the federal register notice pretty much a very extensive discussions on the first principle to radiation protection, health physics and radiological epidemiology.

This is because there are a lot of discussion, this is because the discussions that we're having now are tossing around a lot of numbers. And it may not be apparent to the casual reader what these numbers actually mean in terms of actual radiation health effects.

The range proposed control room design values is significantly lower than the thresholds for observable deterministic health effects such as acute radiation syndrome and hematopoietic syndrome which range around 70 to 100 rad. Which is an acute dose.

Additionally, this range is far below the mean lethal dose of ionizing radiation without medical treatment, which is between 300 and 500 rad. And then lastly, is far below the criteria range for estimating individual estimates of cancer risk mortality given the short time frame of these types of exposures.

We know from the radiation protection and emergency response programs that we actively monitoring actual occupation exposures before the Part 20 limits are met under an actual event. Also, the traditional deterministic DBA consequence analyses will continue to be conservative providing additional defense-in-depth from the Part 20 exposure limit.

And the big takeaway is that all of this really, looking at all of this as a whole and the radiation protection framework, the emergency response framework, looking at the recommendations from these other organizations, and then the real actual health physics underpinnings of these values do ensure a high level protection and still be provided well minimizing long-term health impacts.

We'll go on to Slide 21. Now that we've

established the Commission's radiation protection framework and the evidence-based justifications for a higher numerical value, we can talk about providing some reasonable regulatory relief. Our findings from our work is that adequate radiation protection for public health and safety, and occupational radiological safety can still be achieved at a higher but still safe control room design criteria performance levels while balancing both dose-savings to workers and providing some regulatory relief to maintain operational flexibilities.

We're going to talk about some of the Staff's examinations in regards to just how much margin the light, the fleet has. If we can move on to Slide 22 please.

What we did was we went and looked at as many as we could, the analysis of record, design basis accident radiological consequence analyses, or the entire fleet, that have adopted 50.67. The data set that we have here represents about 84 of these facilities. And what we're looking at here is a histogram of the percent margin of these 84 facilities to the 5 rem control and design criteria itself.

Of these facilities nearly 70 percent of them are within 25 percent of the 5 rem value. So they're about between three and three quarters and 5 rem. And then to a greater degree about 35 percent of the facilities are within 10 rem, 45 to 5 rem of the controlled design criteria value.

Overall the fleet, for the most part, have a relatively small margin to the control design criteria. And we did do a similar analysis on the exclusionary boundary and low population zones. You can see that on Slide 23.

We looked at those same 84 facilities for the exclusionary boundary low population zone. This figure represents their percentage margin to these criteria as well. It is good to acknowledge, as you have mentioned, Member Martin, that the calculations themselves are a little bit different.

So for the control room and the exclusion, the control room and the low population zone, those calculations are done over a 30-day period. And the low population, and the low EAB, or exclusionary boundary, is done over a two-hour period. But what we can generally see here is that there is plenty of margin to these acceptance values for the fleet.

Now let's go on Slide 24. Now that we've given you an idea where the fleet stands in regards to their margin to the control design criteria, I'll present to you our process in designing the proposed rule in developing the regulatory guide.

So, first and foremost, our starting point was considering some Commission directed PRA policies which advocate for certain changes in the development and implementation of its regulations using risk-informed and ultimately performance-based approaches. We have three policies. We have your classic 1985 severe reactor policy statement. We have the PRA policy statement of 1995.

But then more specifically I'd like to point folks attentions to SRM-SECY-98-114. This is a white paper on risk-informed and performance-based regulations. This SECY paper describes the terms and expectations the Commission has for the Staff in developing risk-informed and performance-based regulations.

And we focused on Item 8 within this SRM-SECY. And I'll go ahead and read it now. Stated succinctly, a risk-informed, performance-based regulation is an approach in which risk insights, engineering analysis and judgment, including the principles of defense-in-depth and the incorporation of safety margin and the performance history are used to, one, focus attention on the most important activities, two, establish objective criteria for reevaluating performance, three, develop measurable or calculable parameters for monitoring system and licensee performance, four, provide flexibility determining how to meet the established performance criteria in a way that will encourage and reward improved outcomes, and five, focus on the results as the primary basis for the regulatory decision making.

We took this policy and we divided it up into two elements. The first element is in regards to the rulemaking efforts. Elements, Element 1 took sub-items 2 and 3 to develop the proposed rule, and the Element 2 took Item 4 to develop the guided framework.

And so if we go on to Slide 25, this shows you how we executed this proposed rule for breaking down the SECY into two fundamental elements. Element 1 in a rulemaking we designed a rule which establishes objective criteria for evaluating performance which developed a measurable or calculable parameters from under insistent and licensee performance.

Then Element 2 is the guidance to design a regulatory framework which provides flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes.

So with that in mind, and understanding how the Staff proposed this rulemaking between the two elements we're going to talk about both elements here on Page 26 and 27. Page 26 discussing, or Slide 26 discussing the Element 1.

We're proposing an increase with the control design criteria by a factor of two from 5 rem to 10 rem up to 25 rem with a consideration of plant-specific profiles. These numerical values are risk-informed based on Commission policy and regulation for infrequent and emergency exposures as I have discussed before.

These recommendations from national and international organizations responsible for radiation protection standards and guidance. And then as well as a modern understanding of radiation protection, foundational health physics principles and radiation epidemiology.

We are retaining the performance-based aspect of this rule, which has always been maintained in guidance which requires the traditional design basis accident, radiological consequence analyses to be performed thereby retaining significant and licensee experience in performing these analyses, at least has to go back 70 years now at this point.

But we're introducing the ability to have some flexibility with the rule and scalability with the rule. Incentivizing safety improvements, which where additional margin may be needed above 10 rem, by taking into consideration the way the facility has been designed and operating. That could allow for a higher, but still safe, performance criteria.

Let's go on to Slide 27, please. This is Element 2 of our approach to proposing the submitted rule.

So we're establishing in guidance a framework for a graded risk-informed and performance-based control room design criteria in part based off of comments received from regulatory bases document.

The intent of this framework is to enable a performance-based evaluation using traditional deterministic radiological consequence analysis methods within defined risk-informed boundaries.

These boundaries are defined by acceptable radiation exposure guidelines of radiation workers during accident and emergency conditions and acceptable nuclear risk profiles using modern PRA methods.

The intent of this framework is to provide some flexibility when determining how to meet the established acceptance criteria in a way that encourages and rewards the facility's safe profile.

In practice, the framework will leverage its safe design and operation to justify a higher numerical value for a lower plant-specific risk metric.

Considerations in developing this framework is that it follows condition policy. It's simple to understand and use, it's leveraging already well-understood nuclear safety analysis methods, such as DBA consequence analyses and PRA methods.

The approach is similar to other graded regulatory methods such as seen in Reg Guide 1.174. The Standard Review plan also has a graded dose-based acceptance of criteria approach for the exclusionary boundary and low population zone that range from 2.5 rem up to 25 rem as well as some of the more modern uses of frequency consequence curves that we're starting to see now with the advanced LWRs and I believe a couple of non-LWRs as well.

We did develop a White Paper trying to get this thought out into, you know, the ether so people can take a look at it, digest this method, understand its bases, its rationale. You can find that White Paper at the bottom of the slide here, the ML number at the very bottom.

Okay, so we'll go on to Slide 28. On Slide 28 we're just going to briefly discuss here how we're implementing this framework into DG-1425. The proposed rule, Rev 2 of Reg Guide 1.183.

We ultimately settle on, of course, this graded approach. We base this off of the state of PRA, a review of the state of PRA, the various risk metrics, and a suitability assessment. All of this is described in that White Paper.

We settled on the use of the CDF, or core damage frequency, risk metric to be applied to this framework. A risk metric range is based off of those presented in 1.174 as they are well established and understood with the fleet.

The range of acceptance values goes from 10 rem up to 25 rem. What we are looking at here on the bottom left-hand side of this slide is a screenshot of DG-1425, Table 8, that has this range of acceptance criteria that correlate to a range of CDF bins.

So the intent is that a licensee could take their site-specific -- If they want to go above 10 rem take their site-specific CDF value based off of the acceptable PRA, typically we would, we'll talk more about this tomorrow, accept a PRA value, CDF values for licensing actions that have already been approved.

So if you had done a TSTF-515 type licensing action you could say, hey, look, you know, we've done this PRA work, it's already been approved for this TSTF-505 program, here's our CDF value, if you want to get more information in regards to its technical adequacy and other items, whether or not they close out all the facts and observations from peer review, utilize Regulatory Guide 1.200 for technical adequacy, you can find that there in that licensing action.

So we are trying to leverage that work to inform going up to a higher control design criteria here in this Regulatory Guide.

For those that appreciate, you know, visual figures, you can see here on the right-hand side of this slide where we just simply plopped Table 8 into a nice X/Y chart, or X/Y graph, where if you had a CDF for instance that was higher than 1E to the negative 4 it would be retained in the 10 rem bin, but if you fell into any of those other bins then you could have a higher control design criteria value.

The only way we are utilizing the CDF

value is really only as a sliding scale to identify a higher control design criteria. We're not asking licensees to update their PRAs or to even develop PRAs to implement this.

MEMBER PALMTAG: This is Scott Palmtag.

MR. DICKSON: Yes.

MEMBER PALMTAG: Can you tell me how you came up with these bins on that, was there some --

MR. DICKSON: Regulatory judgement.

MEMBER PALMTAG: It just seems like, and this is my judgement --

MR. DICKSON: Yeah.

MEMBER PALMTAG: -- 1E to the minus 4 seems pretty low so it seems pretty easy to get higher than ten.

MR. DICKSON: Yeah. That range is based off of the criteria that is in Regulatory Guide 1.174. So that guide has a range to make licensing-based changes from 1E to the negative 4 to 1E to the negative 5, other way around, 1E to the negative 5 to 1E to the negative 4, and you utilize that Reg Guide for that framework when you are making or looking at proposed changes at the facility. We just used that same range in this.

MEMBER PALMTAG: Is that the same, all the four ranges are in the Reg Guide?

MR. DICKSON: The max, the min and the max, and then we divvied it up just a little bit further to get a little more detail between 10 rem and 25 rem, and it made for nice bins, bins of 5 rem from 10 to 25 rem.

MEMBER PALMTAG: It doesn't seem that difficult to get up to 25.

MR. DICKSON: So tomorrow we'll talk about that.

MEMBER PALMTAG: Okay.

MR. DICKSON: We have some -- We've done some analyses in regards to where we think the fleet will fall out in this framework and what we did was is we looked at all the different types, I wouldn't say all of them, but quite a few different types of risk metrics, CDF, LERF, full body that used to be used for these dose calculations, even some risk metrics such as population dose risks that are used in like the environmental reports for license renewals, we looked at all these different risk metrics.

We gathered from approved licensing actions the numerical values from those approved licensing actions.

So, for instance, in a TSTF-505

application they will have in their CDF value, they'll have their LERF value, and we created datasets for each of these risk metrics.

We then performed an analysis where we filtered each of the licensees into these bins, developed a histogram to see where they fall out, so we were able to assess based on the quality of the PRA or the intent of the PRA for whatever, you know, regulatory action they were looking at, where they would shake out in applying this.

I'll cut to the punchline, but --

MEMBER PALMTAG: Yeah.

MR. DICKSON: Okay. And we'll talk about this tomorrow, what we ultimately settled on is that we would like them to have PRAs that meet the technical adequacy expectations of a TSTF-505.

That's a high level, that's a high bar in regards to the development of that PRA and it goes through peer review and we have good confidence in that value and allow them to use a TSTF-505 program for risk-informed completion times for their limiting conditions of operations and tech specs.

If you use those PRA values none of them received can be in the 25 rem bin.

MEMBER PALMTAG: You'd get -- What's the

average?

MR. DICKSON: I have a histogram. Well --

(Simultaneous speaking.)

MEMBER PALMTAG: Well --

MR. DICKSON: -- I'm just wondering --

MEMBER PALMTAG: -- you'll have more details tomorrow --

(Simultaneous speaking.)

MR. DICKSON: I will have more details tomorrow.

MEMBER PALMTAG: Okay.

MR. DICKSON: The average for the most part -- We can go -- We can add it to tomorrow's presentation.

MEMBER PALMTAG: Yeah.

MR. DICKSON: Most of them stayed within 10 and 15 rem, probably a few of them would get 20 rem, but none of them would benefit from 25 rem within this framework.

I don't quite have an average. I am looking at the --

MEMBER PALMTAG: That's good enough.

MR. DICKSON: Yeah.

MEMBER PETTI: Elijah, even AP1000?

MR. DICKSON: Well, you popped my bubble.

(Laughter.)

MR. DICKSON: So I didn't -- Member Petti, thank you.

MEMBER PETTI: Sorry.

MR. DICKSON: I suspect that the AP1000 would probably fall into that 25 rem category.

MEMBER PETTI: Yeah.

MR. DICKSON: When we did this work they weren't up and running yet and I didn't go and pull that information. I was really focused on the operating fleet that adopted 50.67. That's was the focus at that time.

MEMBER ROBERTS: Elijah, I was wondering why you didn't use LERF as another criterion. The reason I ask is that it's theoretically possible that the degradation control room happened, then it would show up in LERF --

MR. DICKSON: Yeah.

MEMBER ROBERTS: -- if it impeded the ability of the operators to take certain actions that are credited, which is I'd say ideally because we heard last year that Level 2 HRA is kind of in its infancy and there's not a lot of fidelity and actually, you know, modeling the likelihood of operator take this kind of an action --

MR. DICKSON: Yeah.

MEMBER ROBERTS: -- is this all goes to qualitative aspects of a LERF calculation and you're going to see if there is a, you know, significant credit given for operators in a control room who may or may not be challenged to be there --

MR. DICKSON: Right.

MEMBER ROBERTS: -- at the time of this change. So, anyway, I just wondered if you could answer why you didn't LERF and then maybe the second question is if you were to see something like, you know, a qualitative credit for a control room operator, you know, action in preventing a release?

MR. DICKSON: Yes. Thank you for that comment. I thought and prepared for that comment. So we did look at all these different risk metrics in that White Paper.

We didn't go into like extensive analysis as to why we proposed one over another other than CDF. It was several lines of reasoning why we chose CDF, ultimately in the Regulatory Guide.

One, primarily it is the main risk metric that's used for the fleet right now. Then going back to how we derived the in-containment source term, those are based off of CDF analyses in deriving the in-containment source term, so what they do is they run the PRAs, they bin the accident sequences, and develop the Level 2 and MELCOR models that are representative of those accident sequences that make the majority of the CDF calculation.

Some of those discussions are like in NUREG-1465, so we felt that there were some like commonality behind the degradation of that CDF source term, or, sorry, the use of CDF with the in-containment source term and then using it in this area, and then also we went and we took a look at some of our SPAR models, internal SPAR models that we have for PRA, and you can see for some of the BWRs, for instance, that they do need to take some actions in regards to protecting against LERF and we did provide some language in the guide that can qualitatively leverage that type of information to just make sure that people are thinking about fast accidents that happen that can not only melt the fuel but could then challenge containment integrity and see if they are thinking about those things as well.

That language is in the guide. I can't remember it off the top of my head though.

MEMBER ROBERTS: Yeah, thanks. We had a discussion of the Subcommittee last year about the

Level 2 HRA and the ability potentially at some point in the future to be able to model things like operability influence on that preventing some of these scenarios.

MR. DICKSON: Yeah, mm-hmm.

MEMBER ROBERTS: So I'm glad you've got something in the Reg Guide that has at least that thought process.

MR. DICKSON: That's right. Thank you. Okay, so I think we're on Slide 29 now. So just to recap our approach for this rulemaking presenting an amended control room design criteria higher than five is that we focused on three areas.

First we discussed the foundational aspects of the conditions policy and regulation of the radiation protection framework established in Part 20. With an understanding of this framework we were able to assess how a proposal to the control room design criteria could be possible.

Second, after understanding if the proposal fits within the Commission's framework we reviewed evidence-based justifications by these various national and international organizations responsible for radiation protection recommendations that have strong scientific and technical underpinnings.

Third, lastly, we discussed about providing some reasonable regulatory relief by proposing an increase by a factor of two from 5 rem to 10 rem and provide some flexibility up to 25 rem based off of policies provided by the Commission to the Staff to implement those regulations in a performance-based risk-informed manner.

In closing, on Slide 30, the NRC recognizes the challenges licensees face in retaining margin within their licensing basis for operational flexibility purposes and the small amount of margin that we have in the control room design criteria itself.

Now with currently enriched and burned fuels and then going into looking at higher enrichments, ATF designs with higher enrichments, higher burnups to achieve the LOCA cycle times and even eventually some of these power references as well.

One of the key drivers behind this proposal to the amended control room design criteria is to facilitate increased regulatory efficiency and consistency while still continuing to provide adequate protection of public health and safety.

As I mentioned before, as we were starting to look at this as NEIMA had came out back in 19--, what was it 2019, we were starting to hear from vendors and from some other licensees that they were looking to deviate from regulatory guidance based off and due in part from the conservatisms baked into that guidance, but then as well as to the control room, the margins for the control room design criteria itself.

This proposed rule and supporting guidance does execute SRM-SECY-98-144 which defines the terms and expectations the Commission have for developing risk-informed and performance-based rules.

We do believe that this rulemaking effort will also support increased consistency with the Commission's comprehensive radiation protection and emergency response frameworks by realigning the numerical value of the control room design criteria to a potential reactor accident with exceedingly low probability, which is effectively what the in-containment source term represents.

With that, I'll take questions.

CHAIR BALLINGER: Questions from Members?

(No audible response.)

CHAIR BALLINGER: I can tell you my personal opinion is that you guys buried this. Very good.

MR. DICKSON: Thank you.

(Pause.)

CHAIR BALLINGER: Which is why there are no questions, I guess.

MR. DICKSON: We looked at in as many ways as we possibly could. We appreciate that.

CHAIR BALLINGER: Well if there aren't any questions that concludes the presentations for the day. We should probably thank you again because you got us about two hours ahead of time.

Now we have a period of overall, I guess overall discussion. You folks don't have to sit up there and monitor us.

MR. DICKSON: Okay.

CHAIR BALLINGER: We have a kind of an unusual situation with respect to the letters that we would write in that the letter is due in February and we have Subcommittee meetings now and we have a Subcommittee meeting in January where we'll complete the presentations related to the rule and it's not until, in theory, after that that we would write a letter.

That's a fairy long period of time, so I am hopeful that reasonably quickly, we've had a lot of discussion, detailed and back and forth, that Members would consider providing me with some thoughts of key pieces that we should include in the letter.

There is a lot of -- We've had to have to review an awful lot of material, not all of which is in any controversial, and so we should probably try to stick with stuff in any letter that we would write that would be related to safety or policy, I guess.

We had a lot of discussion this morning related to the FFRD issue, so I'm suspicious, I'm suspecting that we'll get a lot of comments there, but that's kind of the -- We'll have the transcript within three days.

Weidong is all over this, and so everybody, including the people that are virtual, will be able to read the transcript. So hopefully we can enshrine some of this a little bit earlier so we don't forget. I don't forget, for sure. So what do folks think?

MEMBER PETTI: So just in general, Members, we have two days in January.

CHAIR BALLINGER: Yeah.

MEMBER PETTI: We only have one Reg Guide, okay, but the reason we're retaining it is because it seems like we need to spend some time deliberating as a committee on this.

The graph and depth of this is difficult. It's probably one of the most significant rulemakings in terms of impact on the industry, and so I feel like we need to show we've done our due diligence with it in terms about the letter sense.

It would be nice if we could get sort of a consensus before the January meeting and then just hash out any small details, but I just don't know how we'd do that.

CHAIR BALLINGER: I'm kind of hoping that the input that I would get I would, without violating our rules with respect to deliberation and FACA and all that stuff, that something like an interim something.

MEMBER PETTI: Interim points, yeah.

CHAIR BALLINGER: Interim points that would be produced. So that's a starting point.

MEMBER PETTI: But we could look at that in January and --

CHAIR BALLINGER: Yeah. Yeah. If we wait too long we'll lose track.

MEMBER PETTI: Yeah, because then we only have one week --

CHAIR BALLINGER: Yeah.

MEMBER PETTI: -- for Ron to finalize the letter in front of Full Committee.

CHAIR BALLINGER: Yeah.

MEMBER HALNON: Hey, Ron?

CHAIR BALLINGER: Yes, sir?

MEMBER HALNON: This is Greg. Just two quick comments.

CHAIR BALLINGER: Oh, wait a minute, should we let the court reporter go --

MEMBER PETTI: Yes. Do the public comments first.

MALE PARTICIPANT: Public first.

MEMBER HALNON: Okay, I'll hold off.

CHAIR BALLINGER: Oh, that's true, yes. Before we continue let me get the public comments.

MEMBER HALNON: Okay.

CHAIR BALLINGER: If there are members of the public that would like to make a statement could you please state your name and organization, I suppose, and then make your comment.

(Pause.)

CHAIR BALLINGER: Oh, I'll get to you.

MR. CSONTOS: Okay.

CHAIR BALLINGER: I'm waiting for the radio to, for the radio to come back in here.

MR. BURKHART: Yeah, this is Larry Burkhart from the ACRS Staff. If anybody online would like to make a comment please raise your hand and we'll take your comments sequentially and then we'll take comments from the --

(Simultaneous speaking.)

CHAIR BALLINGER: That's what I was going to do next.

MR. BURKHART: So if anybody online interested in making a comment please raise your hand.

CHAIR BALLINGER: Well something happened.

MR. BURKHART: Paul Clifford.

CHAIR BALLINGER: Ah.

MR. BURKHART: Paul Clifford, give us your comment, please.

MR. CLIFFORD: Yes. Hello. I appreciate the opportunity to make a comment. This is Paul Clifford, a consultant with Framatome.

I would like to make a comment on backfit and how it fits into the draft rule package. So the draft 50.46(a) rule is defined as a voluntary alternative to the existing 50.46 ECCS performance requirements and is being pursued to support future licensing actions involving high enrichment and burnup.

Section VIII(f) of the Statement of Considerations provides a backfit assessment for the draft 50.46(a) rule and in that backfit assessment the Staff concludes that holders of existing operating or combined licenses would not be required to comply with the proposed amendment and would have the option to continue their current treatment of LOCAs.

Therefore, the proposed revisions would not meet the definition of backfitting in 10 CFR 51.09(a)(1) and would not constitute backfitting for an operating license or affect the issue of finality of a combined license.

For a majority of the currently operating fleet their "current" treatment of LOCAs does not consider fuel fragmentation and fuel dispersion, the hydrogen-enhanced beta-layer embrittlement, breakaway observation or clad ID oxygen diffusion.

Furthermore, their definition of LOCA is limited to breaks in piping. The Staff's -- It's important to note that the Staff's designation of this rule as a voluntary alternative is a departure from its classification of the 2016 50.45(c) draft rule which was defined as adequate protection and as an adequate protection rule that earlier draft included a mandatory implementation of the research findings.

Given the low safety significance associated with large piping breaks that we've heard over the last two days, the industry supports the voluntary alternative designation of this rule and the backfit assessment.

In accordance with the backfit assessment though the industry expects that these new or modified Staff positions will not be fully fit on licensing actions not involving high burnup or high enrichment, such as a potential wave of future power uprate LARs. That's it. Thank you for the opportunity.

CHAIR BALLINGER: Thank you. Now comments -- Well I haven't seen anybody else online. Members of the public in the room?

MR. CSONTOS: Thanks, Ron. Thank you very much. We do have, the industry has a few comments that we would like to make, just some high-level discussion points.

First of all, we want to thank the Staff for all of the hard work. In fact, this Friday marks three years when the SECY went up to the Commission for the increase in reg rulemaking and then followed by April of that year, of the next, the following year, being the one where the SRM comes down, you know, to move forward with the enrichment rulemaking.

Then this past April the other SRM that came in to incorporate, which was this (a) and (c), into the rulemaking. So there has been a lot of activities and a lot of effort and we thank the Staff for all of the things that they've done.

One thing that we would like to, just for me to highlight is that one of the things that we have been seeing as an overarching kind of a general comment is we see individual groups, technical groups, having different and varying levels of risk-informing or various levels of prescriptive and deterministic and sometimes risk-informed approaches to various activities.

We're looking to see if there is an overarching group that might, and maybe it's ACRS, but there is an overarching group that looks at this holistically and looks at how you implement which are being put out there in the rule, because part of this is that we in the industry really want this rulemaking to be completed because we have strategic aspirations for high burnup, high enrichment, high performance cycles for PWRs as well as power uprates.

Applications are coming in as early as this coming, early first quarter of next year, okay, on some of this, and so having regulatory stability and predictability is really important going forward.

On this case we see differences in different technical disciplines where there are different approaches being taken, and so maybe there is a way to ameliorate some of those inconsistencies maybe or an approach to looking at it in a more risk-informed overarching manner that can help us with moving forward more quickly.

I know it's a challenge. I know it's hard. I know we don't have a lot of time between now and March, so maybe that's something that we take on, you know, further as we go with this process, but it's just something that we just wanted to bring up, that's all. I'll turn it over to Jim.

MR. STAVELY: Jim Stavely, PSEG Nuclear. First of all, thank you. The comments and the presentations over the last two days has been really informative and I do appreciate a lot of the insightful comments and questions asked by the ACRS Subcommittee, so thank you.

I would like to reinforce the comment on enabling the implementation of this planned guidance, but we've heard some things over the last couple days that may present perhaps a few challenges on how we could implement this in the timeframe we are hoping for for the strategic changes of whether it's SBU or whether it's 24 month fuel cycles.

As Al explained, some of it appears to be a little bit of coordination or integration and we can do better perhaps and look forward to a containment conversations and a few other things to make sure we have a smooth implementation of this rule and without minimizing any unexpected impacts that we didn't foresee on implementation, so appreciate that and make that comment.

Also in terms of 50.46(c), just a little bit of a concern. Again, this is a first look and I understand fully why the Staff and management took the position of moving the proposed 50.46(c) from a few years ago with minor changes into this new rulemaking and new guidance documents.

But I think with that approach we perhaps made a little bit of a differential between some of where the current risk-informed performance-based manner was not perhaps as prevalent when we saw 50.46(c) before.

So I would just recommend or consider if there is things in there we need to take a little more of a look at of how we move into to 50.46(c).

Last, ALS. We didn't hear that much about

ALS, the Alternate Licensing Strategy. The industry and Salem puts a lot of emphasis behind the risk-informed nature of that.

That really penalizes some of taking that type of manner and applying it to our current knowledge base and our current expertise and leveraging that for strategically moving the industry forward in a safe manner.

I would just appreciate the continued emphasis on ALS. Thank you.

CHAIR BALLINGER: Other comments?

MS. GERKEN: I'll go real fast. I just want to add on to what he's saying. So I just want to -- Lisa Gerken, Framatome.

So just adding on to what Jim was talking about about 50.46(c) and even the dispersal stuff as well, I think when you are looking at it, you know, straight and flat it can -- You know, we talked about a little bit earlier about the picture, it makes it seem really significant when operationally it's not that significant.

So we have been operating with the 17 percent limit, for instance, with the 50.46 and then you look at the hydrogen-based curve and the limit comes down.

That can present a picture of, wow, there's a big concern, but it's not that we're not accounting for degradation in these phenomena, we're just accounting for it in a different way.

So the risk may be perceived as higher because of, you know, how it appears on a picture or in discussions, but maybe it's less significant than it actually is.

The same thing with dispersal, we do operate today in ranges where dispersal could happen and terminology like "no dispersal" is a little bit misleading as well as in terms of the scariness of the consequences, I think, that means, oh my God, it explodes and stuff is going to go everywhere, you know, coming back to where LOCAs, how they look, actually happen and the physics of them.

The physics of fragmentation and dispersal are important considerations when promulgating new regulation, especially stuff that can be enclosed in rather conservative ways.

We don't want to make it more difficult to do stuff that we're already doing today, both with respect to 50.46(c) and dispersal. Thank you.

MR. SMITH: Thank you. Fred Smith from EPRI. The discussion and the large amount of angst about dealing with dispersal consequences is very reminiscent of discussions that EPRI had when we sought to develop and ALS strategy.

We elected to not take a strategy that would require evaluation of dispersal. What I didn't hear in your discussion was I think a very important element, that any realistic accident sequence for a large break LOCA has to include leak rate protection.

Leak break detection at current expects would require a plant to shut down. Some of you were here in June when I presented the ALS framework and we said and still say that our probabilistic fracture mechanics analysis and large break piping demonstrate that the time between the detectable leak and LOCA was only, was 19 months on the 95/95 vessels.

And so the detectability has a large margin in it and, of course, if you shut down you lose stored energy, decay heat, and you lose the motive force for a LOCA and if a LOCA were to occur it would have no consequences on the cladding rupture or dispersal. Thank you very much.

MR. KINDRED: Good afternoon. Tom Kindred, Southern Nuclear. Can everyone hear me?

(No audible response.)

MR. KINDRED: Okay.

MR. BURKHART: Can you just spell your name so we get a true record?

MR. KINDRED: Yeah. Kindred, K-I-N-D-R-E-D. My comments are in regard to NUREG-1829. The industry, we agree, that NUREG-1829 seems to be applicable.

But what we're seeing, at least from the presentations yesterday, is that as our understanding of, quite frankly, the mechanics pictures and comparisons, you know, up to 2024 with xLPR, we're seeing decreasing trends of high rupture frequencies.

So we are a little concerned within the industry and we don't understand how with an increase in trend of frequencies we're going to mandate more frequent plant inspections and that's going to provide real dose to our nuclear workers, like those inspections will require nuclear workers to get real dose, right, for a hypothetical event.

So that's one thing we don't understand how a decrease in risk we're going to get a risk-informed recommendation to increase the frequencies for risk, yeah.

MR. BROADBENT: I think I'm the last. I am Greg Broadbent, a consultant to NEI. I just wanted to address something that was just mentioned about leakier control rooms.

If you look at the release fractions from Rev 0, which most of the plants are on right now, to Rev 2, a lot of them have doubled. Some of them have gone up even more that.

So, you know, by going to this higher enrichment we would expect our doses to approximately double and that's why, you know, we would need an increase in the acceptance criteria.

It's not to make our control rooms leakier, it's to be able to not make, have to make wholesale changes to other things to the plant.

MR. BARBER: Kevin Barber, Westinghouse. I want to talk about the true best estimate definition, or maybe the lack thereof.

I mean the Member made a comment yesterday about how you can get a very different result, whether it's the 50th quantile of the LOCA population or whether you apply a LOCA methodology using nominal models as is.

You know, the Staff had talked about it and certainly the Westinghouse letter in response to the reg basis being noted that we think that there is a good probability that we can demonstrate, you know, rupture with the true best estimate methodology.

That is with a caveat though. There are a number of aspects or a number of models that we need to ensure are being treated realistically, for example like the break size.

That is something that is very important. If we are looking at the break size and we decide that even though it's a beyond design basis accident we're going to just look at the double-ended guillotine breaks instead of a realistic distribution of the probability of breaks, that's a significant bound.

That is the type of requirement that, you know, whether it gets codified or whether it gets in the Reg Guide that we need to make sure as an industry that we have clarity and efficiency in that to ensure that we can deliver on what we're saying about a true best estimate really not having, leading to feel this way.

MR. PORTER: One comment. Ian Porter, GNF. So just a quick comment on the 50.46(c) aside the new Reg Guide. So when that was first created, you know, ten years ago there were a number of cladding alloys that were actually approved and in use at the time.

Since that was written there have been three new alloys that have been approved and they are actually in use or in the state of deployment and they are not included in the Reg Guide updates as approved alloys that can actually use the Reg Guide.

So it's not clear if those approvals are going to actually require the vendors go do the testing for the new alloys that have been approved the last ten years or if there is applicability.

So just a comment that there is not clarity with what's actually happened with new alloys that are currently licensed after actually use of the Reg Guide would have to go be tested.

CHAIR BALLINGER: There are probably 50 people in the room and we've already had 30.

(Laughter.)

CHAIR BALLINGER: Okay. If there aren't any other public comments now I think we can release the court reporter. Thank you. Thank you.

(Whereupon, the above-entitled matter went off the record at 2:35 p.m.)

Proposed Rule: Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors

December 17-18, 2024

Opening Remarks

Theresa Clark
Director
Division of Safety Systems

Overview of Increased Enrichment Rulemaking

Philip Benavides
Project Manager
Reactor Rulemaking & Project Management Branch

Issue Identification

- **Regulatory Issue:**
  - Current licensing framework allows for the use of ≤ 5 weight percent uranium-235; however, technology developments may require numerous exemptions to utilize fuel enriched above 5 weight percent.
- **Proposed Solution:**
  - Rulemaking would provide for a generically applicable standard informed by public input, providing consistent and transparent communication, rather than individual licensing requests as discussed in SECY-21-0109, "Rulemaking Plan on Use of Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors."
- **Commission Rulemaking Plan Approval:**
  - Staff request to pursue rulemaking and develop a regulatory basis was approved by the Commission via SRM-SECY-21-0109.

SRM-SECY-21-0109 Overview

- SRM-SECY-21-0109 was issued on 3/16/22, in response to SECY-21-0109.
  - The Commission approved the staff's proposal to initiate a rulemaking to amend requirements for the use of light water reactor fuel containing uranium enriched to greater than 5.0 weight percent uranium-235.
  - Provisions to the rule should only apply to High-Assay Low Enriched Uranium (HALEU).
  - Fuel Fragmentation, Relocation, and Dispersal (FFRD) should be appropriately addressed.
  - Staff directed by the Commission to take a risk-informed approach.

Status of Rulemaking Activity

- **The NRC staff issued a regulatory basis on September 8, 2023 (ADAMS Accession No. ML23032A504)**
- **Stakeholder Involvement:**
  - Before Regulatory Basis Issued:
    - Public Meeting on June 22, 2022 (ML22208A001)
  - Post Regulatory Basis Issued:
    - Public Meeting on October 25, 2023 (ML23319A259)
    - Comment Period closed on January 22, 2024
    - Publicly shared Fuel Dispersal insights at the NRC's Annual Higher Burnup Workshop on September 3, 2024 (ML24277A161)
- **The Increased Enrichment proposed rule package is in concurrence.**
  - Proposed rule due to the Commission: March 2025

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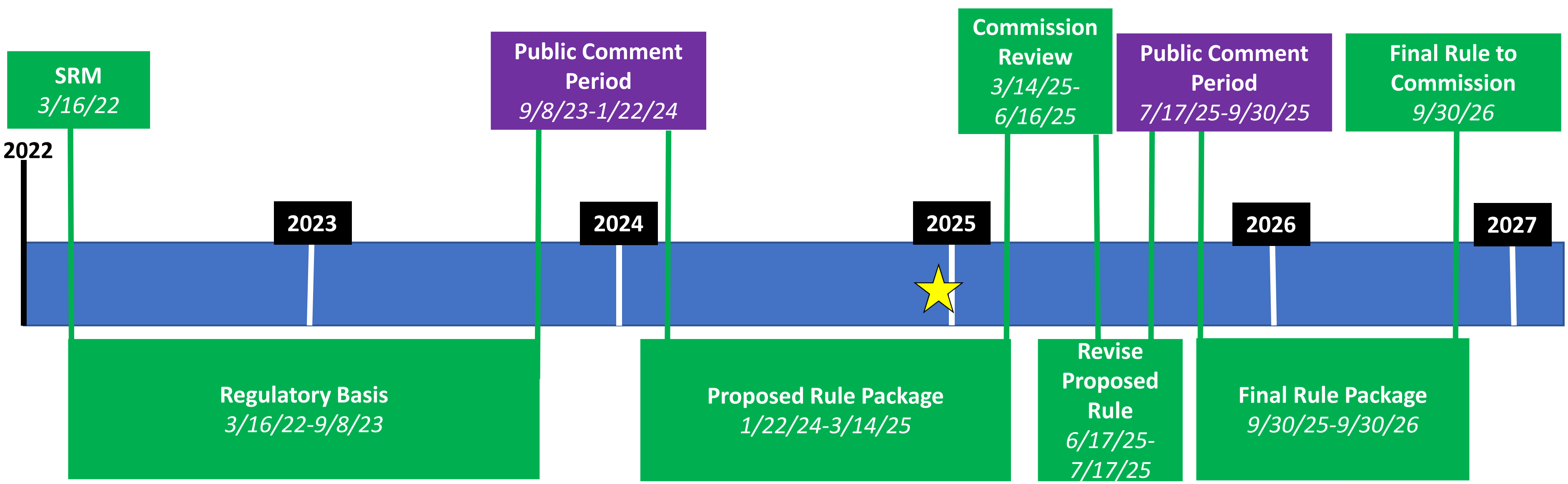
Status of Rulemaking Activity



Note: Dates listed are estimates only, and thus are subject to change.

Rulemaking Topics

- **The IE rulemaking addresses the following topics:**
  - Criticality Accident Requirements (10 CFR 50.68)*
  - Uranium Fuel Cycle Environmental Data - Table S-3 (10 CFR 51.51)
  - Environmental Effects of Transportation of Fuel and Waste - Table S-4 (10 CFR 51.52)
  - Packaging Requirements for Fissile Material Transportation (10 CFR 71.55)*
  - Control Room Design Requirements (10 CFR 50.67 and GDC-19)*
  - Fuel Fragmentation, Relocation, and Dispersal*

*ACRS Subcommittee Meeting Topics

Associated Guidance

- **Control Room Design Requirements (10 CFR 50.67 and GDC-19)**
  - DG-1425, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”
- **Fuel Fragmentation, Relocation, and Dispersal**
  - DG-1261, Revision 1, “Measuring Breakaway Oxidation Behavior”
  - DG-1262, Revision 1, “Determining Post-Quench Ductility”
  - DG-1263, Revision 1, “Establishing Analytical Limits for Zirconium-Based Alloy Cladding”
  - DG-1426, “An Approach for Risk-Informed Evaluation Process Supporting Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Reactors”
  - *DG-1428, “Plant-Specific Applicability of Transition Break Size”*
  - DG-1434, “Addressing the Consequences of Fuel Dispersal in Light-Water Reactor Loss-of-Coolant Accidents”

Note: DG-1428 to be presented in January 2025

Agenda (December 17)

Topic	Presenter	Organization
NRR Staff Leadership	Theresa Clark, Director	NRR/DSS
IE Overview, Status of Rulemaking	Philip Benavides, Rulemaking PM	NMSS/REFS/RRPB
Criticality Accident Requirements (10 CFR 50.68)	Charley Peabody	NRR/DSS/SNSB
Fissile Packaging Requirements (10 CFR 71.55)	Jason Piotter	NMSS/DFM/NF
Fuel Fragmentation, Relocation, and Dispersal	Joseph Messina Robert Tregoning David Rudland Kristy Bucholtz	NRR/DSS/SFNB RES/DE NRR/DNRL NRR/DRA/APOB

Agenda (December 18)

Topic	Presenter	Organization
NRR Staff Leadership	Theresa Clark, Director	NRR/DSS
Fuel Fragmentation, Relocation, and Dispersal (Continued)	James Corson	RES/DSA/FSCB
Control Room Design (10 CFR 50.67 and General Design Criteria 19)	Elijah Dickson	NRR/DRA/ARCB

Questions

Criticality Accident Requirements (10 CFR 50.68)

Charley Peabody
Nuclear Engineer
Nuclear Systems Performance Branch

10 CFR 50.68(b) Historical Overview

- Final Rule issued in 1998
- Provides alternative to 10 CFR 70.24
- Implements k-effective safety limits
- Limits enrichments of ≤ 5% by weight U-235
- Industry desires to increase enrichment beyond 5% at operating LWRs

10 CFR 50.68(b) Recent Developments

- Single License Amendment for Lead Test Assemblies (LTA) utilizing enrichments beyond 5% has been approved to date
- NRC Contracted Research Study
  - ORNL/TM-2024/3350 “Scoping Studies on the Impacts of Increased Enrichment on Nuclear Criticality Safety,” May 2024 (ADAMS Accession Number ML24163A016)
- Reconfirmed Regulatory Basis Decision not to Update Regulatory Guide 1.240 “Fresh and Spent Fuel Pool Criticality Analysis” March 2021

10 CFR 50.68(b) Proposed Rule Unofficial Redline

- §50.68(b)(7) would be changed to state "The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight or to the value specified in the operating license."
- No changes to any other paragraphs.

10 CFR 50.68(b) Proposed Rule Benefits

- The k-effective safety limits specified in 10 CFR 50.68(b)(2), (3), and (4) are maintained
- Increased enrichment criticality safety impacts will be evaluated as part of the fuel transition LAR review process
- Allows for the entire range of high-assay low-enriched uranium (HALEU) to be utilized
- This alternative preserves 10 CFR 50.68(b) compliance for all LWRs currently operating in the United States

10 CFR 50.68(b) Proposed Rule

Member questions or comments for the staff?

Packaging Requirements for Fissile Material Transportation 10 CFR 71.55

Jason Piotter
Team Leader - New Fuels Team
Division of Fuel Management - NMSS

Packaging Requirements of 10 CFR 71.55: Summary of Regulatory Issue

Current Regulations

- § 71.55(b) applicants evaluate a single package, optimally moderated and reflected
- § 71.55(g) Provides an exception to § 71.55(b) for packages containing UF_6
- § 71.55(g)(4) Specifies that enrichment of UF_6 cannot exceed 5 weight percent U-235

Regulatory History

Proposed rule (§ 71.55(g)) issued 67 FR 21390, April 30, 2002, Final Rule issued 69 FR 3698, January 26, 2004

Codified NRC longstanding practice to provide an exception to § 71.55(b)

10 CFR 71.55: Certificate of Compliance (CoC) Options for Enrichments Greater than 5 weight percent U-235

1

Evaluate UF_6 packages with optimum moderation per §71.55(b), or

2

Request an exemption to §71.55(b), or

3

Request approval under §71.55(c) for an exception to the optimum moderation requirement in §71.55(b).

Regulatory Basis - Rulemaking Alternatives

Original Recommendation →

Option 1: No Action	71.55(b), 71.55(c), exemptions (Also applies to Options 2 and 3)
Option 2: Increase Enrichment Limit	20 weight percent U-235
Option 3:	Remove Enrichment Limit

Proposed Rule - Rulemaking alternatives

Current Recommendation → Option 2

Option 1: No Action	71.55(b), 71.55(c), exemptions (Also applies to Options 2 and 3)
Option 2: Increase Enrichment Limit	10 weight percent U-235, *defense in depth requirements*
Option 3: Increase Enrichment Limit	20 weight percent U-235, *defense in depth requirements*

10 CFR 71.55(g)(4): Updated Recommended Alternative

Alternative 2

Increase enrichment limit to 10.0% wt U-235, with prescriptive defense in depth requirement for additional protection of fill valve or other device

10 CFR 71.55(g): Updated Rule Text

(4) The uranium is enriched to not more than *10 weight percent* uranium-235; and

(5) A design feature is incorporated to protect the valve or other fill device from impact for contents with uranium-235 enriched above 5 weight percent and up to 10 weight percent.

Fissile Material Transportation Packages - Summary

- Transportation packages for UF_6 have a certification pathway with enrichments up to 20 weight percent U235. While our current regulations as written are sufficient to transport higher enriched UF_6, we are providing a non mandatory modification of the current enrichment limit that allows for more regulatory certainty while maintaining safety

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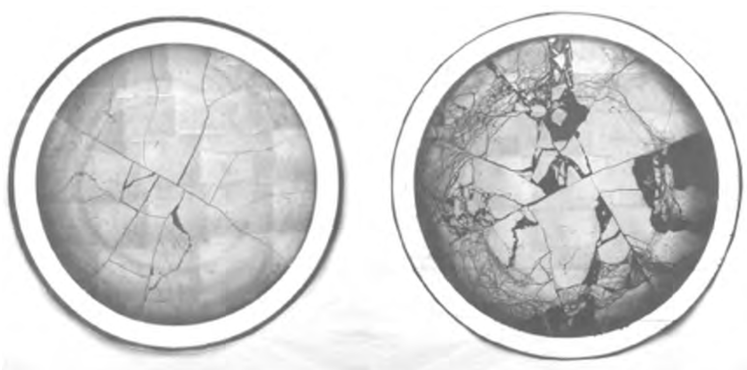
Fuel Dispersal and 10 CFR 50.46a Overview

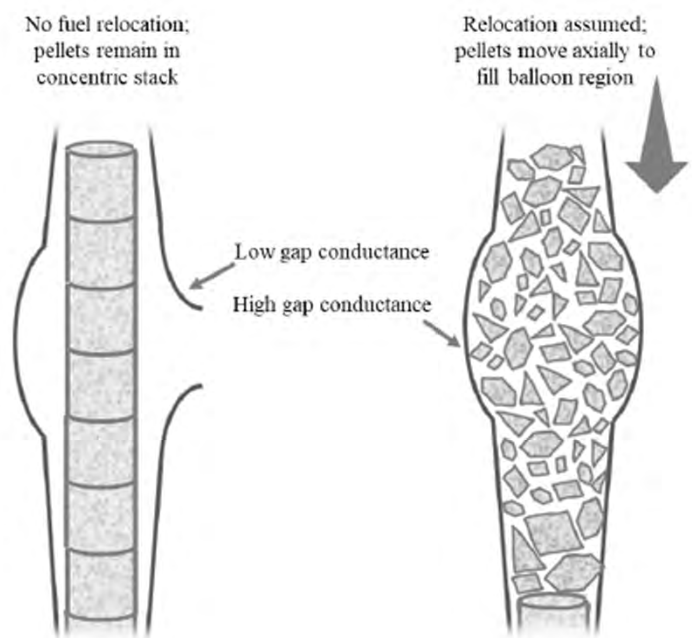
Joseph Messina
Reactor Systems Engineer
Nuclear Methods and Fuel Analysis

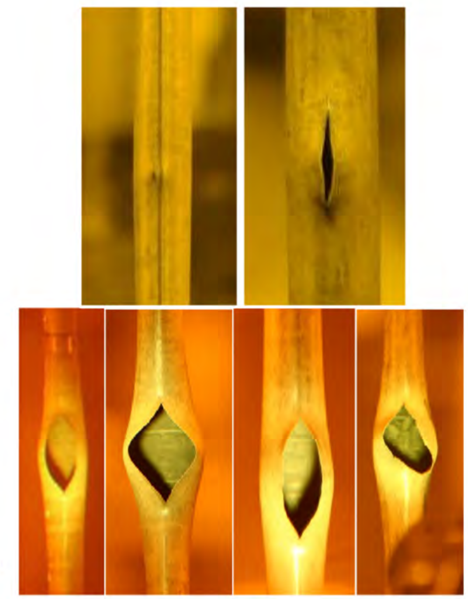
Fuel Fragmentation, Relocation, and Dispersal (FFRD)

- At HBU experiments have shown that the fuel can fragment during a LOCA
- Differences in pressure across the cladding can lead to cladding ballooning and burst
- The fragmented fuel can relocate axially into the balloon region of the fuel rod and if burst occurs, disperse into the RCS



Segment from NRC's ANL LOCA program at 55 GWd/MTU before and after testing





Burst openings from Studsvik LOCA tests (NUREG-2121)

Fuel Dispersal: Background and Regulatory Issue

- The 50.46 acceptance criteria date to 1974 when FFRD were not known phenomena
- Acceptable approaches to demonstrate compliance with the regulations have ensured that catastrophic failure of the fuel rod structure and loss of fuel bundle configuration are precluded
  - Fuel dispersal would be a departure of precedent
- Fuel dispersal is not explicitly addressed within the current regulations
  - Draft proposed rule language allows for some flexibility regarding fuel dispersal
  - DG-1434 provides guidance for addressing fuel dispersal within the proposed rule

IE Rulemaking Regulatory Basis FFRD Alternatives

The IE Rulemaking Regulatory Basis (ML23032A504) considered 5 licensing pathways for addressing fuel dispersal:

- **Alternative 1**: No action.
- **Alternative 2**: 50.46a-style modification of ECCS requirements.
- **Alternative 3**: Perform a safety demonstration for post-FFRD consequences.
- **Alternative 4**: Provide a generic bounding assessment of dose and use risk insights for post-FFRD consequences.
- **Alternative 5**: Use probabilistic fracture mechanics to show that leaks in large pipes will be identified before failure, precluding the need to analyze LBLOCAs.

Public Comments Overview, Part 1 of 4

- No unanimous alternative recommended by industry
- Industry recommendations were strongly based on the qualitative schedule impacts published in the Regulatory Basis.
  - These were quick and crude estimates.
  - Staff has learned more, and accuracy of these estimates has improved.
- UCS and two members of public:
  - Do not support any alternative that allows for fuel dispersal
- One member of public:
  - Recommends waiting until more research and analysis is performed for fuel dispersal

Public Comments Overview, Part 2 of 4

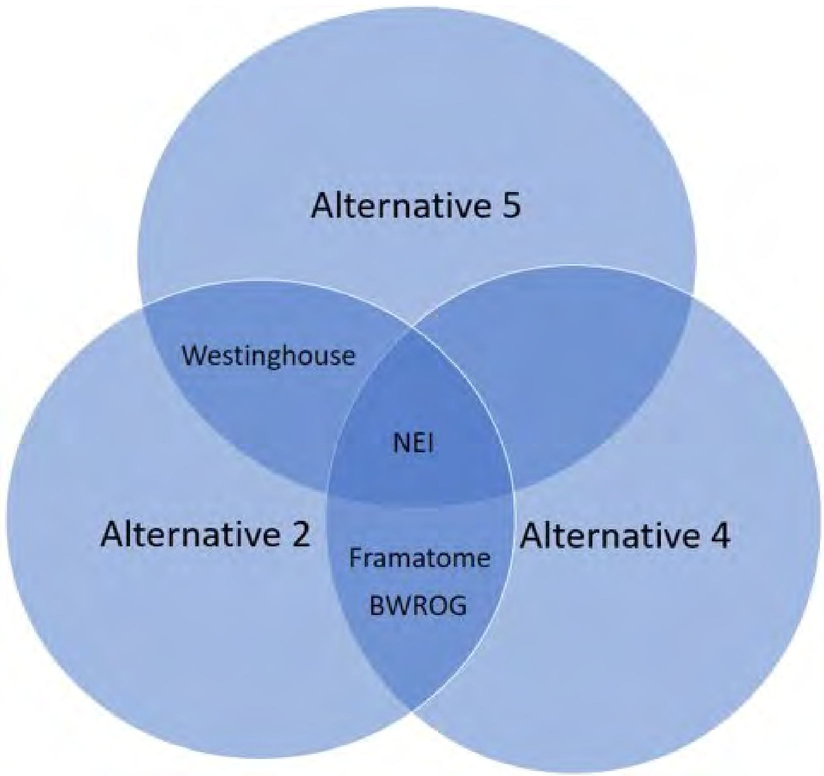
Alternatives 1 and 3:	Alternative 4:	Alternative 5:	Alternate Approaches:
• No support	• NEI and BWROG supported Alternative 4 • BWROG does not see Alternative 5 as a solution	• NEI and Westinghouse supported a modified Alternative 5 (ALS)	• Framatome and PWROG suggest using integrated decision making as done to disposition in-vessel downstream effects (IVDEs) associated with GSI-191

Public Comments Overview, Part 3 of 4

Alternative 2:

- NEI, BWROG, and Westinghouse support Alternative 2 combined with an updated 50.46c as a separate rulemaking due to perceived schedule or as a backup
- Aspects of Framatome's response and their 2023 white paper align with Alternative 2
- NEI, Westinghouse, and Framatome stated that even with a no-dispersal criterion, Alternative 2 would be reasonable with true best-estimate calculations

Public Comments Overview, Part 4 of 4



Previous 50.46a History

- **Commission SRMs on rule development**:
  - SRM-SECY-02-0057 (ML030910476)
  - SRM-SECY-04-0037 (ML041830412)
  - SRM-SECY-05-0052 (ML052100416)
  - SRM-SECY-07-0082 (ML072220595)
- **ACRS letters**:
  - ACRS letter on initial draft final rule (ML063190465): 11/16/2006
  - ACRS letter on NUREG-1829 and draft NUREG-1903 (ML073440143): 12/20/2007
  - ACRS letter on draft final rule (ML102850279): 10/20/2010
    - Sept. 2010 ACRS Subcommittee meeting transcript: ML102910759
    - Oct. 2010 ACRS Full Committee meeting transcript: ML102860120
- **SECY-10-0161 (ML102300252)**: Draft final rule was submitted to the Commission – 12/10/2010
- **Email from Greg Bowman to SECY** requesting to withdraw 10 CFR 50.46a rulemaking (ML121500380; submitted in response to verbal direction by Chairman Jaczko) – 4/20/2012
- **SRM-SECY-10-0161**: Commission approved staff's request to withdraw SECY-10-0161 and re-evaluate it to ensure compatibility with future Commission direction related to recommendations following Fukushima – 4/26/2012
- **SECY-16-0009 (ML16028A189)**: staff recommended stopping 10 CFR 50.46a rulemaking as a part of prioritization and re-baselining of agency activities – 1/31/2016
- **SRM-SECY-16-0009 (ML16104A158)**: Commission approved staff recommendation to stop 50.46a rulemaking – 4/13/2016

Fuel Dispersal Path Forward

- The NRC staff plans to risk-inform LOCAs by modernizing 50.46a (based on Alternative 2) in the IE draft proposed rule to facilitate addressing fuel dispersal
  - Support for Alternative 2 expressed in many public comments
  - Smallest impact on the IE Rulemaking schedule of the alternatives that received support
    - Leveraged the technical basis and work performed in the original 50.46a
  - High level of technical maturity
- 10 CFR 50.46a was a draft final rule in 2010 that proposed to establish a transition break size (TBS), above which LOCAs would be recategorized as beyond-design-basis
  - Voluntary alternative to 50.46
  - Original philosophy being maintained with some changes
  - LOCAs below the TBS will not be affected by this rule
- The updated 50.46a is planned to include high-level, fuel technology neutral, performance-based Emergency Core Cooling System (ECCS) acceptance criteria

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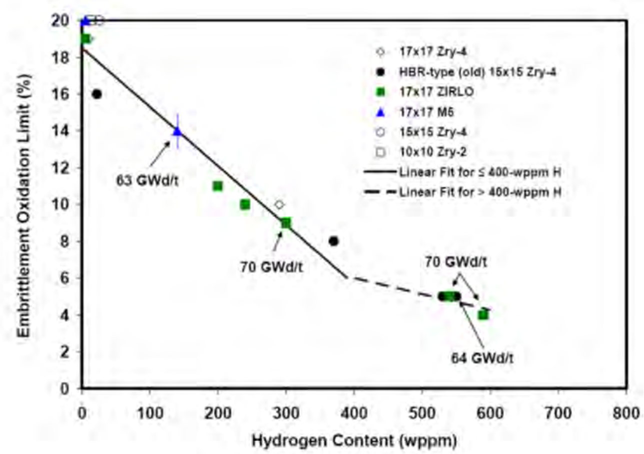
Addressing Fuel Dispersal

- The 50.46a approach is expected to facilitate safety demonstrations of fuel dispersal because true best-estimate modeling and realistic assumptions are expected to significantly reduce or eliminate the potential for fuel dispersal
- While this approach does not explicitly address non-mechanistic approaches to evaluating FFRD, as described in other alternatives in the IE Regulatory Basis, other licensing pathways exist
  - E.g., the topical report review process
  - The performance-based criteria are expected to provide relief to the prescriptive philosophy of the existing regulatory framework (including a less prescriptive definition of core coolability)

50.46c Background

- 50.46c was a draft final rule that revised the ECCS acceptance criteria to be performance-based and reflect research findings on embrittlement of zirconium alloy cladding under LOCA conditions
  - Submitted to the Commission via SECY-16-0033 in March 2016
  - 1997-2016 NRC LOCA research program is documented in NUREG/CR-7219
- Substantial ACRS interactions on 50.46c:
  - ACRS letter on draft final rule issued February 2016: ML16048A522
- The SECY-16-0033 research findings show that under the current regulations (17% MLO and 2200°F PCT), post-quench ductility is not assured following a postulated LOCA
- New embrittlement mechanisms discussed in SECY-16-0033:
  - Hydrogen-enhanced beta layer embrittlement
  - Cladding ID oxidation
  - Breakaway oxidation



SRM-SECY-16-0033 (50.46c)

The Commission returned the 50.46c draft final rule package (SECY-16-0033) to the staff in April 2024 without Commission action and directed the staff to do the following:

1. The staff should apply an appropriate risk-informed regulatory approach to address the research findings on cladding embrittlement effects under LOCA conditions described in SECY-16-0033.
2. The staff should evaluate Item 1 with other associated technical issues being addressed, such as fuel fragmentation relocation and dispersal, and risk-informed treatment of LOCAs, including the draft final 50.46a that had been provided in SECY-10-0161.
3. The staff should evaluate whether specific emergency core cooling system criteria such as cladding temperature should be codified or instead addressed in regulatory guidance.
4. Within six months of the date of this SRM, the staff should provide, through a Commissioner Assistant's Note, an action plan for the above items.

(SRM-SECY-16-0033, ML24102A281, April 11, 2024)

50.46c Path Forward

The staff plans to include aspects of 50.46c in voluntary provisions of the Increased Enrichment (IE) proposed rule and assess the need for further action on the 50.46c rulemaking after the Commission votes on the IE final rule package.

- The staff is planning to risk inform LOCAs, as suggested in SRM-SECY-16-0033, with 50.46a in order to facilitate safety demonstrations of fuel dispersal in the IE proposed rule.
- The staff would use the public comments received on the 50.46c aspects of the IE rulemaking to inform any potential future action on the 50.46c rulemaking.
- Entities that elect to adopt 50.46a would be expected to address the embrittlement research findings
- The staff will continue to perform the annual ECCS Safety Assessments and evaluate the impacts of the cladding embrittlement research findings within the framework of existing regulatory requirements when reviewing industry submittals that could result in cladding embrittlement impacts (e.g., power uprates or burnup increases).

Scope of Work Associated with updated 50.46a Proposed Rule

- Confirmation of the transition break size (TBS)
  - NRC internal and external expert elicitation
  - xLPR runs of the NUREG-1829 bases cases
  - Evaluation of operating experience
  - Confirmation of NUREG-1903 technical basis
- Update of the following draft regulatory guides (previously part of the 50.46c rulemaking):
  - DG-1261, "Measuring Breakaway Oxidation"
  - DG-1262, "Determining Post Quench Ductility"
  - DG-1263, "Establishing Analytical Limits for Zirconium-Alloy Cladding Material"
- Development of the following draft regulatory guides:
  - DG-1426, "An Approach for a Risk-Informed Evaluation Process for Supporting Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Reactors"
  - DG-1428, "Plant-Specific Applicability of the Transition Break Size"
  - DG-1434, "Addressing the Consequences of Fuel Dispersal in Light-Water Reactor Loss-of-Coolant Accidents"

Adequate Protection

While this rule relaxes the regulatory treatment of LOCAs above the TBS, the NRC staff believe that it maintains the adequate protection of public health and safety because:

- The initiating event frequency for such events are very low and the NRC will ensure that it is low and remains low on a plant-specific basis
- The NRC will ensure that risk increases from changes due to this rule are minimal and that there are not large increases in the overall plant risk
- The NRC will maintain regulatory control over such LOCAs, continuing to review ECCS evaluation models and plant-specific LOCA analyses, as done to date

50.46a and Fuel Dispersal Team

NRR/DSS:

- Joseph Messina
- Ashley Smith
- John Lehning
- Scott Krepel

NRR/DNRL:

- David Rudland
- David Dijamco
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- Robert Tregoning
- Matthew Homiack
- Christopher Nellis

RES/DSA:

- James Corson
- Andrew Bielen

Technical Basis of Original 10 CFR 50.46a Transition Break Size

Rob Tregoning
Senior Advisor for Materials
Office of Nuclear Regulatory Research, Division of Engineering

Presentation Objective and Outline

- **Objective:** Summarize the historical (i.e., pre-2024) technical basis use to develop the transition break size (TBS)

- **Outline:** TBS development
  - LOCA frequency assessment (NUREG-1829)
  - TBS selection
  - Confirmation of seismic integrity (NUREG-1903)

NUREG-1829: Scope and Objectives

- Develop piping and non-piping passive system LOCA frequencies as a function of leak rate and operating time up to the end of the license extension period (i.e., 60 years) using expert elicitation
  - LOCAs which initiate in unisolable portion of reactor coolant system
  - LOCAs related to passive component aging, tempered by mitigation measures
- Determine LOCA frequency distributions for typical plant operational cycle and history
- Assume that no significant changes will occur in future plant operating profiles

NUREG-1829: Historical LOCA Frequency Evaluation

- LOCA frequencies previously developed from operating history
- Notable Previous Evaluations:
  - WASH-1400 (1975): Estimates largely based on experience in other industries
  - NUREG-1150 (1987): Updated the WASH-1400 distributions to account for the additional service since WASH-1400
  - NUREG/CR-5750, Appendix J (1998): Updated original WASH-1400 study for SB LOCAs while MB and LB LOCA frequencies were calculated from precursor leaks in class 1 systems
- Operating history, by itself, may not accurately reflect future performance and requires significant extrapolation for MB and LB LOCA frequencies

NUREG-1829: Expert Elicitation Process

- Classical approaches
  - Operating experience: LOCA events are rare
  - Plant modeling: Number and diversity of possible failure modes is too complex to accurately model
- Expert elicitation is a formal process for providing quantitative estimates for the frequency of physical phenomena when the required data is sparse and when the subject is too complex to accurately model
- Elicitation has been used often at NRC
  - Development of seismic hazard curves
  - Performance assessments for high-level radioactive waste repository
  - Determination of reactor pressure vessel flaw distributions

NUREG-1829: Elicitation Approach

- Conduct pilot elicitation
- Select panel and facilitation team
- Develop technical issues
- Quantify base case estimates
  - Develop quantitative estimates for well-defined piping conditions
  - Quantify non-piping precursors and targeted failure scenarios
- Formulate elicitation questions
- Conduct individual elicitations
- Analyze quantitative results and qualitative rationale
- Summarize and document results
- Conduct internal and external review of process and results

NUREG-1829: Pilot and Elicitation Team

- **Pilot Elicitation**
  - Conducted using 11 internal (NRC) experts with broad knowledge-base
  - Provided interim results for rulemaking development
  - Developed possible framework for subsequent elicitation and its strengths and weaknesses
  - Identified technical issues for subsequent consideration

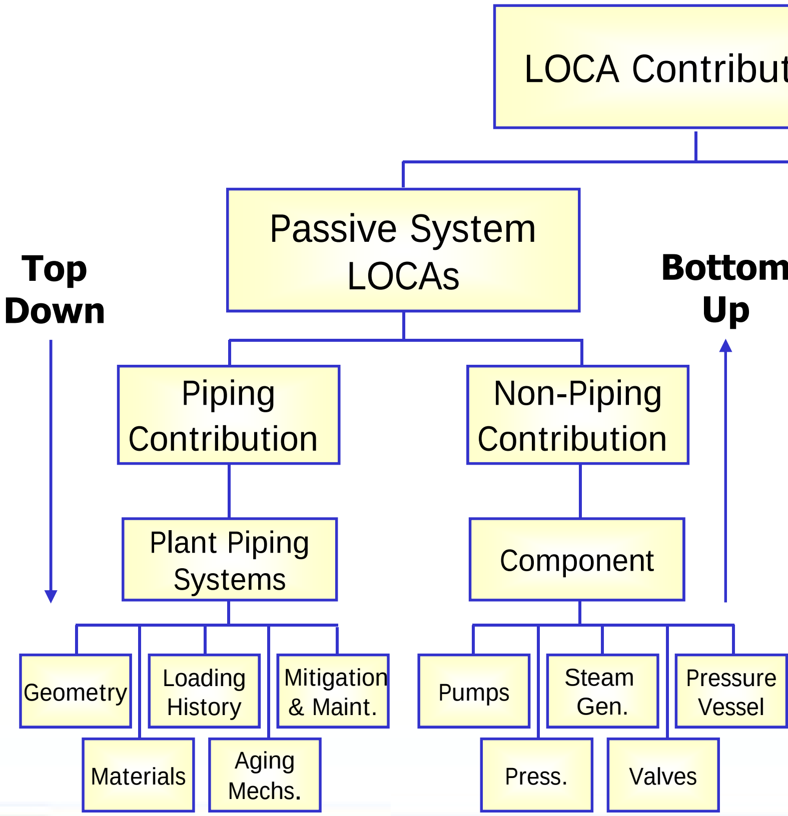
- **Panel and Facilitation Team**
  - Individual elicitations conducted for each expert, led and monitored by a facilitation team
  - Twelve external experts assembled from nuclear industry, DOE laboratories, consultants, and international regulatory agencies with broad knowledge-base
  - Facilitation team comprised largely of NRC subject matter experts

NUREG-1829: LOCA Size Classification

- LOCA sizes based on flow rate to group plant system response characteristics
  - First three categories similar to NUREG-1150 and NUREG/CR-5750
  - Three additional LBLOCA categories used to determine larger break frequencies
- Correlations developed to relate flow rate to effective break area
- Three time periods evaluated
  - Current day ~ 2004 (average 25 years of operation)
  - End of design life (average 40 years of operation)
  - End of first life extension (average 60 years of operation)

Category	Flow Rate Threshold (gpm)	LOCA Size
1	> 100	SB
2	> 1500	MB
3	> 5000	LB
4	> 25,000	LB a
5	> 100,000	LB b
6	> 500,000	LB c

NUREG-1829: General Issue Classification

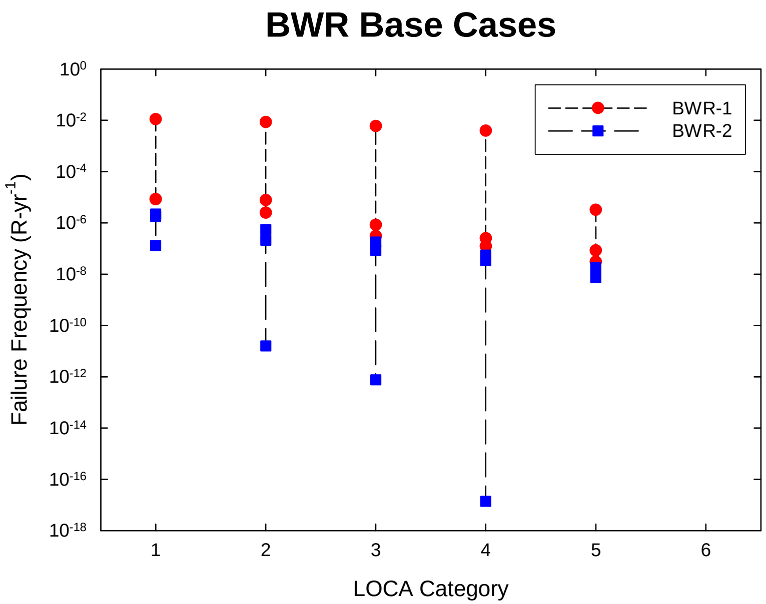


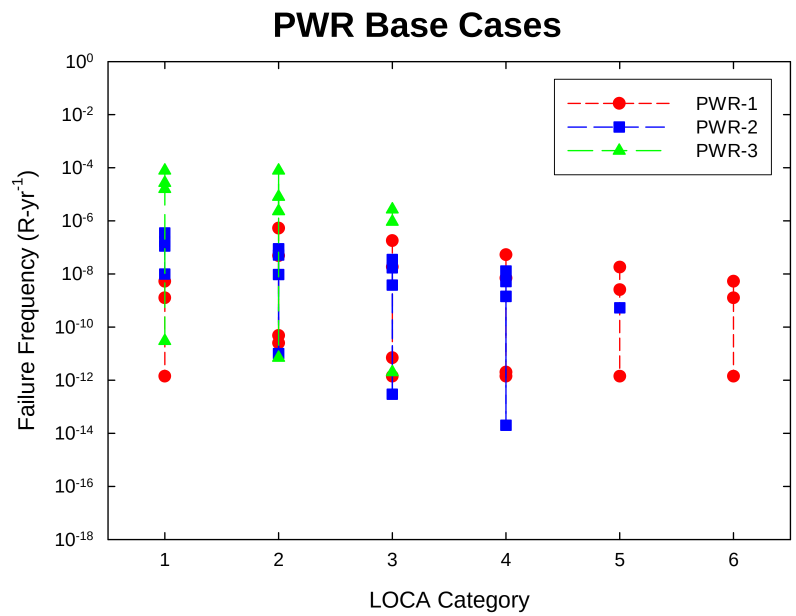
- Elicitation focuses on passive system LOCAs
- Important piping and non-piping attributes identified
- Elicitation structure supported top-down and bottom-up analysis

NUREG-1829: Piping Base Cases

- The base cases were available for anchoring the elicitation responses.
- Base case conditions specify the piping system, piping size, material, loading, degradation mechanism(s), and mitigation procedures
- Five base cases defined
  - BWR
    - Recirculation System (BWR-1)
    - Feedwater System (BWR-2)
  - PWR
    - Hot Leg (PWR-1)
    - Surge Line (PWR-2)
    - High Pressure Injection makeup (PWR-3)
- The LOCA frequency for each base case condition is calculated as a function of flow rate and operating time
- Four panel members individually estimated frequencies: two using operating experience and two using probabilistic fracture mechanics

NUREG-1829: Piping Base Cases Summary Results





- Large variability due to inconsistencies in both the conditions evaluated and differences in approaches
- Each base case participant presented their approach and results to entire panel
- Each panel member was asked to critique approaches & results during their elicitation session

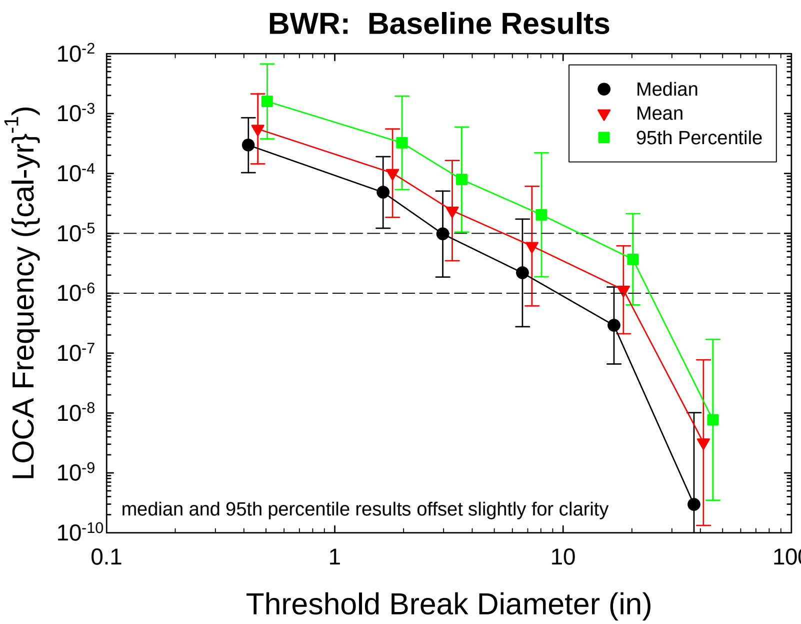
NUREG-1829: Non-Piping Base Cases

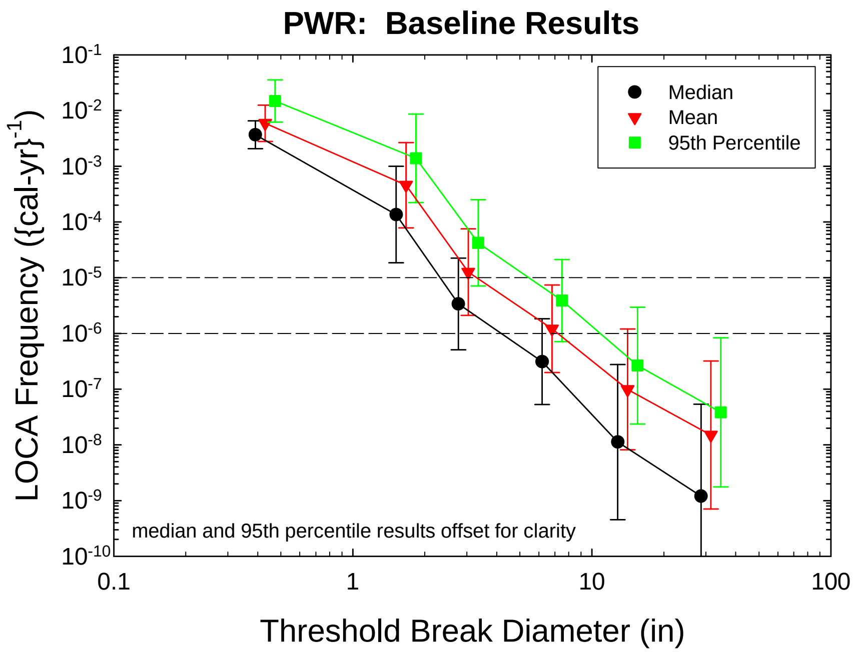
- The variety and complexity of the non-piping failure mechanisms makes the piping base case approach intractable
- Approach
  - Develop general non-piping precursor database
  - Use PFM modeling to develop LOCA frequencies for targeted degradation mechanisms
    - CRDM ejection
    - BWR vessel rupture: normal operating and LTOP
    - PWR vessel rupture: PTS
- Analysis method
  - Choose appropriate base case: non-piping precursor, piping precursor, piping base case, or non-piping base case
  - Determine relative likelihood of each non-piping failure scenario compared to chosen base case

NUREG-1829: Analysis of Elicitation Responses

- Calculate individual estimates for each panelist
  - Total BWR and PWR LOCA estimates
  - Approach is self-consistent and ensures that qualitative rationale and quantitative estimates match
- Aggregate individual estimates: Philosophy
  - Group results more accurate than any single estimate
  - Outliers should not dominate quantitative estimates
- Aggregate individual estimates: Approach
  - Combine parameters (i.e., mean, median, 5th & 95th percentiles) of individual distributions
  - Calculate confidence bounds associated with each parameter estimate
- Perform sensitivity analyses to evaluate calculation approach
- Final LOCA distributions reflect uncertainty and variability
  - Uncertainty: Individual panel member responses
  - Variability: Range of individual responses

NUREG-1829: Total LOCA Frequencies





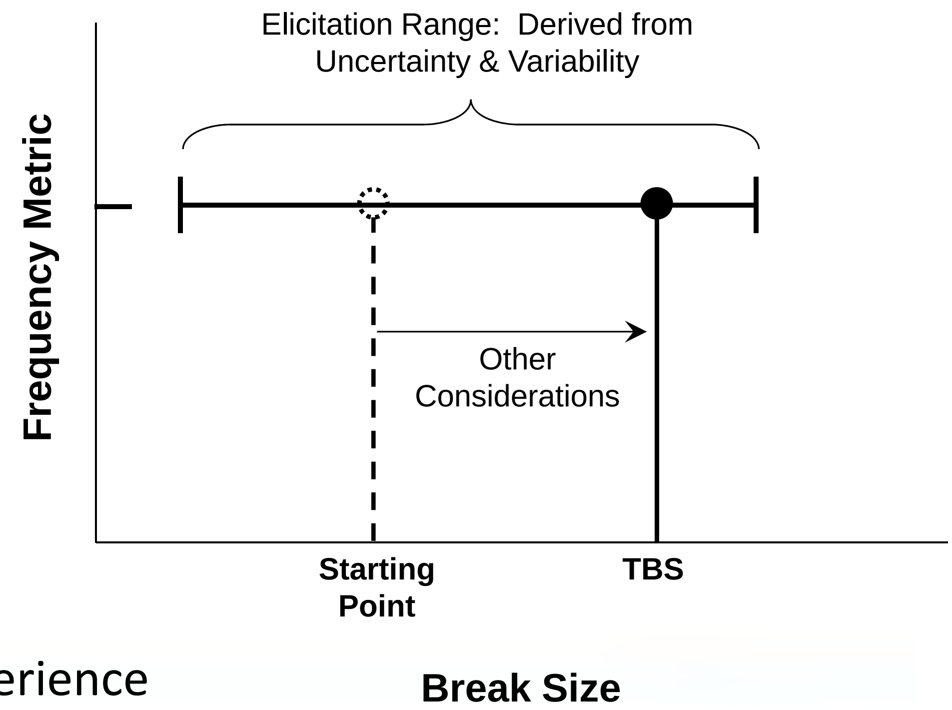
- 95% confidence bounds (i.e., error bars) account for diversity among panelists
- Differences between median and 95th percentiles reflect individual panelist uncertainty

NUREG-1829: Summary

- Formal elicitation process used to estimate generic BWR and PWR passive-system LOCA frequencies associated with material degradation during normal operations
- Piping and non-piping base cases were developed and evaluated for anchoring elicitation responses
- Panelists provided quantitative estimates supported by qualitative rationale in individual elicitations for underlying technical issues
  - Generally good agreement on qualitative LOCA contributing factors
  - Large individual uncertainty and panel variability in quantitative estimates
  - Results are generally comparable to NUREG/CR-5750 estimates
- Group results determined by aggregating individual panelists' estimates
  - Geometric mean aggregated results are consistent with elicitation objectives and results are generally comparable with NUREG/CR-5750 estimates
  - Alternative aggregation schemes can result in higher LOCA frequencies

Selection of Transition Break Size (TBS)

- NUREG-1829 results used as starting point
- Range of pipe sizes correlate to break frequency < 10^{-5}/yr (95th percentile)
  - BWRs: 13 to 20 inches
  - PWRs: 6 to 10 inches
- Selection should accommodate uncertainties
- Other types of LOCAs considered in determining TBS
  - Active LOCAs
  - Load-generated LOCAs (i.e., dropped heavy loads, water hammer)
  - Seismically induced LOCAs
- Actual plant piping design and operating experience considered in final selection



TBS Selection

- TBS is defined as a pipe break that is the size of the cross-sectional flow area of largest pipe attached to the main coolant loop
  - For PWRs, the size of the largest pipe attached to the cold or hot leg main loop piping (≈ 12 inches)
  - For BWRs, the size of the largest pipe in either of the RHR or Feedwater systems inside primary containment (≈ 20 inches)
- Supporting rationale
  - Next larger pipes are significantly less likely to break
  - Piping sizes < TBS have experienced most significant degradation
  - Accommodates uncertainties and provides regulatory stability as variation in future LOCA frequencies estimates not likely to require new TBS definition to maintain acceptable risk

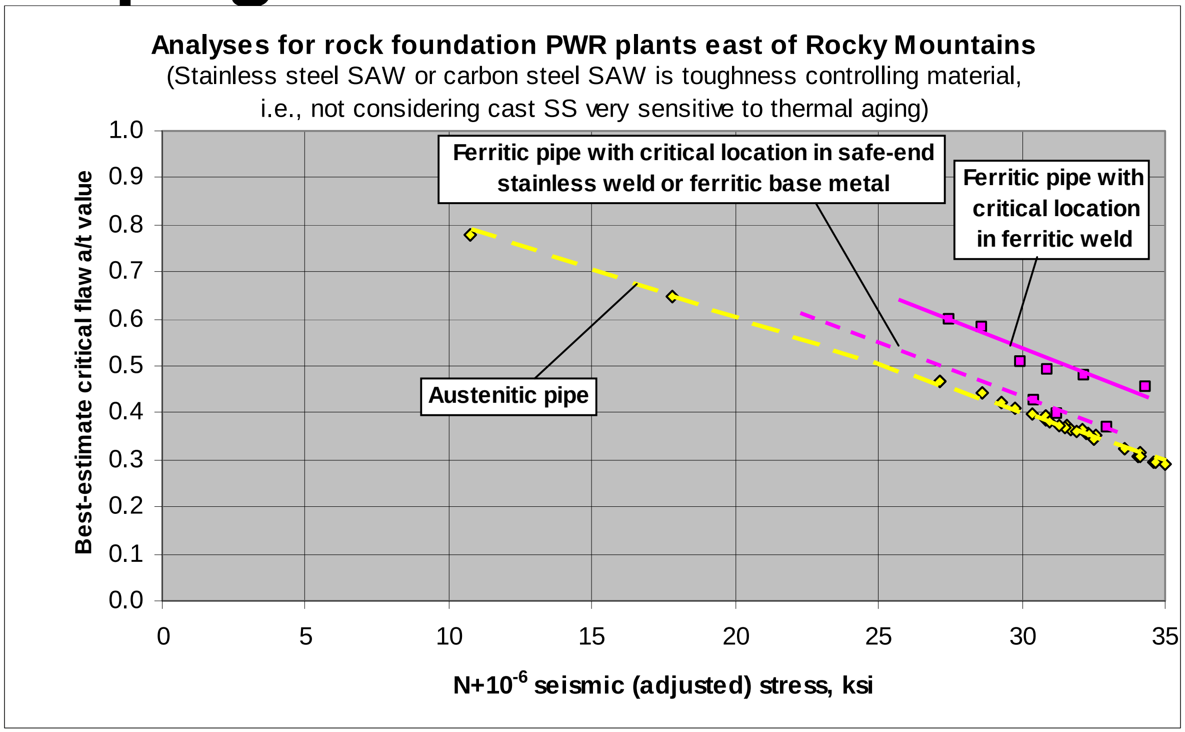
NUREG-1903: Objective and Approach

- **Objective**
  - Determine if seismic risk is acceptable for breaks greater than TBS
- **Scope and Approach**
  - Six supporting activities
    - Unflawed piping failure
    - **Flawed piping failure**
    - **Indirect piping failure**
    - Review of past earthquake experience
    - Review of past seismic PRAs
    - Review a mid-80s LLNL study of direct and indirect seismic piping rupture used to support GDC 4 revision
  - Use mix of deterministic and probabilistic approaches

NUREG-1903: Approach, cont.

- Analyzed direct piping failure under rare seismic events
  - Evaluated unflawed and flawed piping systems with diameters > TBS (e.g., hot leg, cold leg, and cross-over leg) using available design information
  - Used most-recent seismic-hazard curves for plants east of the Rocky Mountains
  - Determined stresses for 10^{-5} and 10^{-6} yr^{-1} seismic event by scaling plant specific SSE stresses
  - Apply scale factors to address conservatisms in the design process, material behavior, and extrapolation to rare seismic loading
- Analyzed indirect piping failure under rare seismic events
  - Analyzed large component support failures that may lead to piping failure
  - Assumed that support failure leads directly to piping failure
  - Updated results from prior LLNL study to reflect new hazard and ground motion information
  - Determined mean failure probability of component supports

Direct Piping Failure: Surface Flaw Results



- 26 PWRs analyzed
- Critical flaw depth (a/t) for long flaw ($\theta/\pi \approx 0.8$) under 10^{-6}/yr seismic event

Indirect Piping Failure: Case Studies

NUREG/CR-3663 Sample Results

Group A Plants **(Combustion Engineering)**	Confidence Limit (1)		
	10%	50%	90%
Calvert Cliffs	2.3×10^{-8}	6.1×10^{-7}	6.1×10^{-6}
Millstone 2	9.0×10^{-10}	6.6×10^{-8}	1.2×10^{-6}
Palisades	5.0×10^{-7}	6.4×10^{-6}	5.2×10^{-5}
St. Lucie 1	1.2×10^{-8}	3.8×10^{-7}	4.1×10^{-6}
St. Lucie 2	6.6×10^{-8}	1.4×10^{-6}	1.1×10^{-5}
Westinghouse Lowest Capacity Plant	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}

(1) Confidence limit of 90% implies a 90% confidence that annual probability is less than value indicated

- Generic seismic hazard curves used in evaluation
- Group A had highest failure probabilities for CE plants

NUREG-1903

- Only 2 plants evaluated
- Mean result for Calvert Cliffs: 1.7E-6/yr

NUREG-1903: Summary

– Unflawed piping: Failure frequency is much lower than 10^{-5}/yr

– Flawed piping

  – Critical flaws for long, circumferential flaws ($\theta/\pi = 0.8$) are generally large
    - 40% of wall thickness for 10^{-5}/yr seismic event
    - 30% of wall thickness for 10^{-6}/yr seismic event

  – Conditional probability of breaks larger than the TBS should be less than 10^{-5}/yr

– Indirect failures

  – Only two cases analyzed (one W and one CE plant)

  – Piping failure induced by major component support failure has a mean probability of approximately 10^{-6}/yr

Historic TBS Technical Basis Development: Summary

- Passive System LOCA frequencies developed for generic BWR and PWR plants through an expert elicitation process (NUREG-1829)
  - Accounted for panelist uncertainty and variability among responses
  - Used results as the starting point for selecting the transition break size
- Increased TBS to address additional factors and to promote regulatory stability
  - Considered other types of LOCAs
  - Accounted for plant piping design and operating experience
- Performed confirmatory study to determine if risk of LOCAs > TBS due to rare seismic was acceptable (NUREG-1903)
  - Risk due to unflawed and flawed direct piping failures expected to be acceptable for most, if not all, plants
  - Risk due to indirect piping failures acceptable for two cases evaluated
  - Seismic risks, however, are plant-specific, making it difficult to completely generalize results

Recent Confirmation of the Transition Break Size Technical Basis

David Rudland
Senior Technical Advisor for Materials
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Office of Nuclear Reactor Regulations

Confirmation Study

- Confirmation of the NUREG-1829 LOCA Frequencies
- Confirmation of the NUREG-1903 Results
- Determination of TBS impact

NUREG-1829 Confirmation

- Internal and External Elicitation
- Impact of Recent Operational Experience
- Probabilistic Fracture Mechanics Study
- International Operational Database Study

Qualitative (Internal and External Elicitation; Impact of Recent Operational Experience)

Quantitative (Probabilistic Fracture Mechanics Study; International Operational Database Study)

Details in "White Paper on Continued Applicability of NUREG-1829"
ML24205A015

Internal and External Elicitation

- **Motivation**
  - NUREG-1829 based on formal expert elicitation
    - Pilot elicitation performed initially
    - External elicitation formulated based on lessons-learned from internal pilot
  - Mimic process to evaluate the completeness and continuing viability of the NUREG-1829 and NUREG-1903 results
- **Objectives**
  - Identify possible scenarios either not considered or under-estimated in NUREGs-1829 and 1903
  - Assess likelihood and/or technical or rulemaking gaps associated with each scenario

Approach

- **Select appropriate internal and external panelists**
  - **Internal**: 13 senior staff with collective expertise in all relevant technical areas
  - **External**: Two NUREG-1829 panelists with complementary expertise pertaining to passive system reliability
- **Formulate initial set of questions and topics**
  - Focus on knowledge gained and operating experience since the mid-2000s
  - Consider direct, indirect, and potential common-cause failure scenarios
  - Identify important causal factors
- **Hold a kick-off meeting**
  - Present objectives, background and motivation of the effort
  - Discuss and clarify the elicitation topics and questions
  - Identify initial considerations
- **Develop initial independent responses**
- **Conduct follow-on meetings**
  - Collectively discuss the individual responses
  - Determine the path forward for dispositioning any open issues

Summary of Internal Elicitation Responses

- **Scenarios – 21 identified**
  - **Addressed in applicability studies**: RPV embrittlement, SCC in main loop piping, increased seismic risk since NUREG-1903, evolution of ISI and relief requests
  - **Addressed within 10 CFR 50.46a rulemaking**: Rulemaking motivation, effects of future plant changes, PRA representativeness, indirect failures from small pipe rupture, TBS margin, treatment of LBB piping, degraded supports and snubbers, NUREG-1829 uncertainties, BWR applicability, maintaining mitigative capabilities, definition of a pipe
  - **Addressed within current regulations:** pilot-operated relief valve failure, common-cause maintenance errors, RPV through-wall cracking, water chemistry excursions, impact on plant security, degraded grid stability

Addressed later

U.S.NRC

Sample of Internal Elicitation Topics

- **Treatment of LBB piping**
  - **Issue**: Consideration of special treatment for plants with LBB approval
  - **Disposition**:
    - No explicit special treatment in rule although NUREG-1829 results reflect LBB margins
    - May be able to leverage approved LBB analysis as part of plant-specific applicability demonstration
- **Potential for degraded grid stability**
  - **Issue:** Higher risk could result if LOOP is not evaluated within LOCA analysis
  - **Disposition:**
    - LOOP event frequencies, while relatively sparse, don't indicate an increasing trend
    - PRA still needs to consider risk associated with such events and continually update data
    - Many plants currently employ load monitoring software to predict offsite power unavailability

Summary of External Elicitation Responses

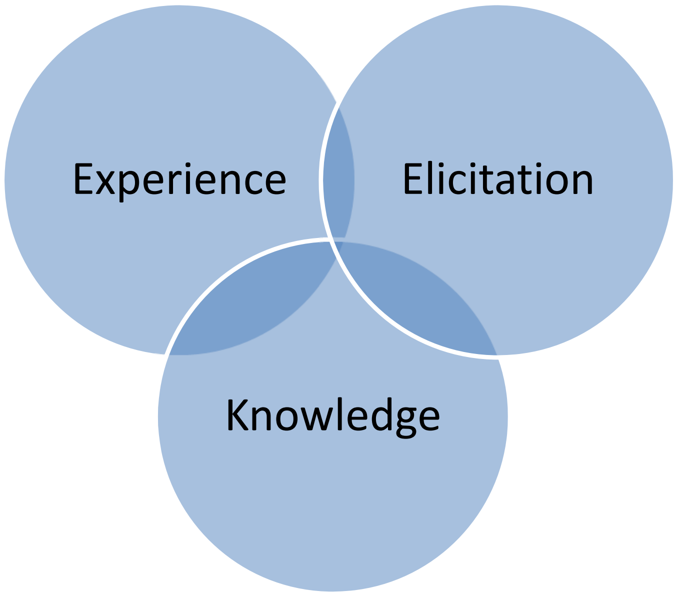
- **Continuing Applicability of NUREG-1829 LOCA frequencies**
  - Frequencies for breaks < TBS are representative (one panelist) or conservative (one panelist)
  - Frequencies for breaks > TBS are conservative
  - Opinions are based on successful mitigation practices and increased knowledge pertaining to the structural integrity of large piping systems
- **Historical TBS remains viable**
- **Possible scenarios leading to breaks > TBS**
  - One panelist: no such credible scenarios envisioned
  - One panelist
    - Should continue to explore thermal aging, cold work, and residual stress effects
    - SCC leading to long, shallow surface flaws especially in either CASS or adjoining weld or cold-worked component are most credible
    - Likelihood of such scenarios is strongly plant-specific

Disposition of External Elicitation Responses

- **Continued research and monitoring**
  - Harvest representatively aged austenitic weld and CASS materials to validate current models, which are based largely on accelerated aging laboratory testing
  - Extend aging studies to represent properties at the end of subsequent license renewal period and beyond
  - Continue to monitor both U.S. and international operating experience relevant to potential for SCC cracking, especially in large diameter piping systems
- **Demonstrate plant-specific applicability of the TBS**
  - Proposed 10 CFR 50.46a rule requires that an entity demonstrate that plant-specific effects do not invalidate the applicability of the TBS for their plant before implementing the rule
  - Additional guidance proposed in DG-1428 providing several methods for demonstrating plant-specific applicability

Impact of Recent Operational Experience (OE)

- Thermal Embrittlement of Piping
- Stress Corrosion Cracking (SCC) of Stainless Steel in PWRs
- Carbon Macrosegregation
- Quasi-laminar Indications
- Small Surface Break Flaws
- Reactor Pressure Vessel Embrittlement
- Inspection Frequency changes
- Secondary-Side Piping Failure



Impact of Recent OE - Piping

Thermal Embrittlement of Piping	SCC of Stainless Steel in PWRs
• Issue – Decrease in fracture toughness of cast authentic stainless steel and austenitic stainless welds – Can this decrease impact failure frequencies • Staff Action – Considered experimental testing, development of aging management, lack of active degradation, ongoing inspections – No safety concern	• Issue – Many cracks identified in Safety injection and residual heat removal system in stainless welds in French Fleet – unexpected SCC – Could this occur in US and may it impact the failure frequencies • Staff Action – Conducted Risk-informed analysis (LIC-504) – Reviewed industry actions – Determined reasonable assurance of integrity

Impact of Recent OE - Piping

Inspection Frequency Changes	**Impact on TBS**
• Issue – Ongoing efforts at ASME to optimize inspection frequencies resulting in less inspections for piping and components – Risk-informed ISI is in place, but some categories may change inspection frequency • Staff Action – Continuing inspection of these welds is essential to the basis supporting the transition break size.	• Conclusion – Some issues were analyzed through our LIC-504 risk-informed process – Some issues were analyzed through research or licensing actions – No impact on the TBS • Staff Action – However, the staff recommends that for those reactor coolant pressure boundary piping whose diameter is greater than the TBS, a 10 percent sample of the welds > TBS is needed each interval – Can leverage existing ISI programs

Minority View on Inspection Requirements

- Piping failure resulting in LOCA > TBS is highly unlikely, but possible
  - Most prominent concern is an SCC-like mechanism that causes a long-surface flaw with slow through-wall growth coupled with toughness decrease due to thermal aging
    - Increases likelihood of break before leak
    - Issue identified in internal and external elicitations
    - Characteristic of flaws leading to ruptures in PFM analyses
    - Flaws which such characteristics have been occasionally discovered (e.g.., Duane Arnold and Penly 1)
  - Such a scenario is plant-specific, not generic
- Performance monitoring, through inspection, of piping with inner diameter greater than TBS provides assurance that failure likelihood remains extremely low
  - Rulemaking utilizes classical approach of defining a risk-informed inspection sample and then performing repeat inspections each ISI interval
  - Minority view recommends choosing a new risk-informed inspection sample every ISI interval to ensure that a greater population of such welds is inspected at least one time during operation

Impact of Recent OE - Vessels

Carbon Macrosegregation (CMAC)

- Issue
  - In early 2015 regions of (CMAC) were discovered in European Pressurized Reactor pressure vessel heads manufactured for a plant in Flamanville, Manche, France
  - Higher strength, lower toughness, may be more susceptible to embrittlement
- Staff Action
  - Conducted risk-informed analysis, considered EPRI and ASN analyses
  - Concluded that the safety significance of CMAC to the U.S. fleet is negligible

Quasi-laminar Indications (QLI)

- Issue
  - In July of 2012, ultrasonic inspections of RPV ring forgings at two nuclear power plants in Belgium revealed thousands of sub-surface, nearly-axial indications
  - Do the many flaws impact RPV integrity?
- Staff Action
  - Reviewed Electrabel PFM evaluation
  - Conducted independent risk-informed evaluation
  - Concluded the potential existence of QLI is not expected to affect structural integrity of U.S. RPVs.

Impact of Recent OE - Vessels

Small Surface Break Flaws (SSBF)

- Issue
  - 2016 ORNL analyses suggested that SSBF can produce a greater through-wall crack frequency than a 1/4T flaw
  - What are the impacts on P-T limits and PTS?
- Staff Action
  - PFM analyses conducted and determined that even though there is an increase in conditional probability of failure, the impact on through-wall crack frequency is minimal.
  - Realistic cooldown transient frequencies and their occurrence frequency was considered.

RPV Embrittlement

- Issue
  - The existing RG 1.99 (and 10 CFR 50.61) embrittlement trend curve (ETC) model may underpredict of RPV embrittlement under the high fluences
  - Licensees are allowed to defer surveillance capsule testing that is intended to confirm embrittlement predictions from the ETC model
- Staff Action
  - Staff developed a risk-informed analysis that suggested the staff's confidence in the integrity of the RPV for certain plants may be impacted
  - Staff proposed a change to the rule in SECY-22-0019
  - Staff waiting on Commission decision

Impact of Recent OE - Vessels

Inspection Frequency Changes

- Issue
  - Through 10 CFR 50.55a(z) many licensees have been granted approval to modify their inspection intervals for RPV, steam generator and pressurizer welds
  - Cumulative effect of these relaxation may impact the TBS
- Staff Action
  - Ensure reasonable assurance of safety
  - Verify appropriate performance monitor occurs within these components

Impact on TBS

- Conclusion
  - Some issues were analyzed through our LIC-504 risk-informed process
  - Some issues were analyzed through research or licensing actions
  - Cumulative effects were considered
- Staff Action
  - Impact of the embrittlement concerns on the TBS be revisited following Commission action on the rulemaking plan.

Impact of Recent OE

Secondary-Side Piping Failure

- Issue
  - Secondary side failure (<TBS), impacts larger piping (>TBS) and increases LOCA frequencies
- Staff Conclusion
  - GDC-4 and SRP 3.6.2 provide reasonable assurance safety is maintained
  - Piping is very flaw tolerant – probability of enough damage to rupture large piping is small
  - Guidance is needed to cover any possible impacts of secondary side failure causing indirect failure of piping greater than the TBS

Probabilistic Fracture Mechanics Study

- Base Cases from NUREG-1829 (2004) re-examined using improved the state of knowledge and PFM modeling capabilities
- LOCA frequencies recalculated for 4 piping systems relevant to transition break size
- Calculations performed using NRC's extremely Low Probability of Rupture (xLPR) PFM code

Case #	Weld location	Pipe Size	Plant Type	Degradation mechanisms
PWR-1	Reactor Vessel Outlet Nozzle	30-inch	PWR	PWSCC
PWR-2	Surge Line	10-inch	PWR	PWSCC
BWR-1a	Recirculation	28-inch	BWR	IGSCC
BWR-1b	Recirculation	12-inch	BWR	IGSCC

xLPR Approach

- General settings
  - 80 year reactor operation time
  - Sufficient sample size to detect a 10^{-6} probability event
  - Leak rate detection enabled

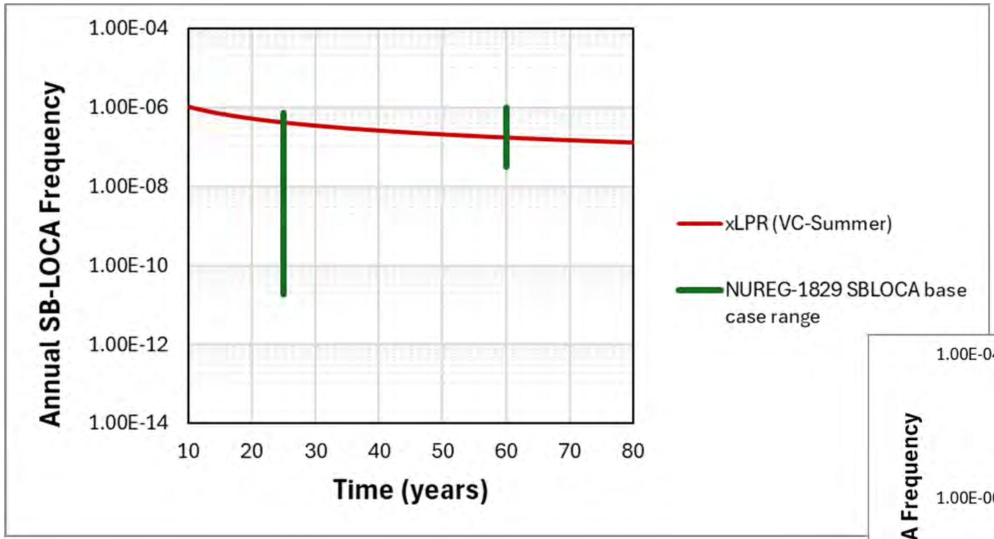
- Quantities of interest
  - Probability of leakage
  - Probability of rupture
  - Leak rate

- Post-processing converts these quantities to annual component-level LOCA frequencies

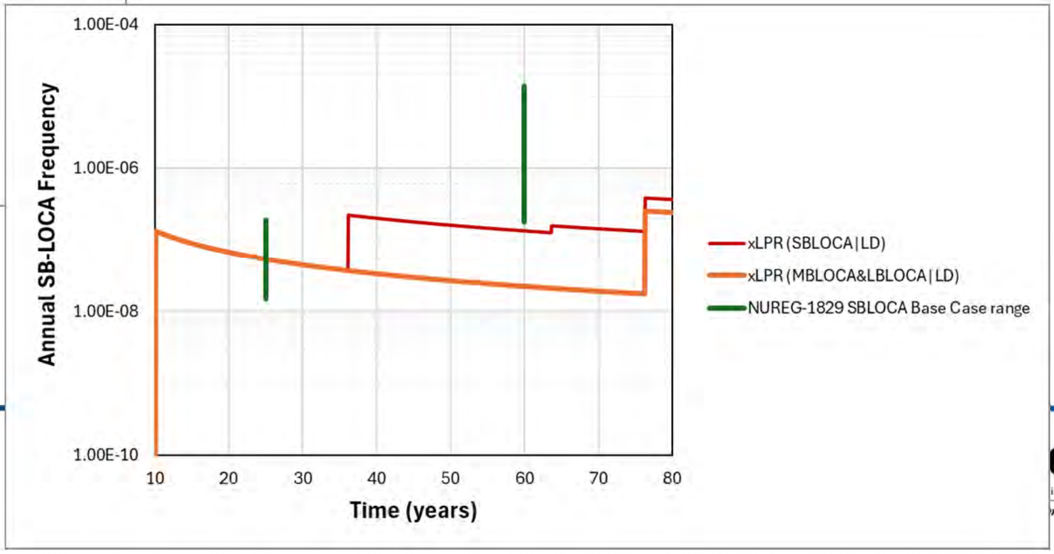
PWR	BWR
PWSCC crack growth	IGSCC crack growth • Generic crack growth model parameters set to match IGSCC model in 2023 ASME Section XI Article Y2310
Weld residual stress • PWR-1: set WRS to mirror profile from VC Summer leak event for conservatism • PWR-2: Generic representative WRS profile	Weld residual stress • Generated using finite element analysis from EPRI data

The PWR probabilistic fracture mechanics analyses were within the range of the NUREG-1829 base case results.

PWR-1: Vessel Outlet Nozzle

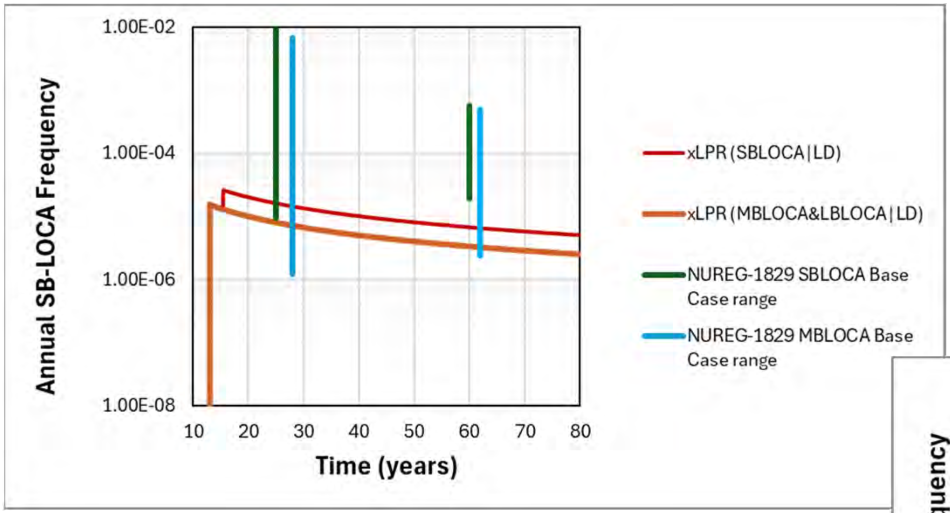


PWR-2: Pressurizer Surge Line

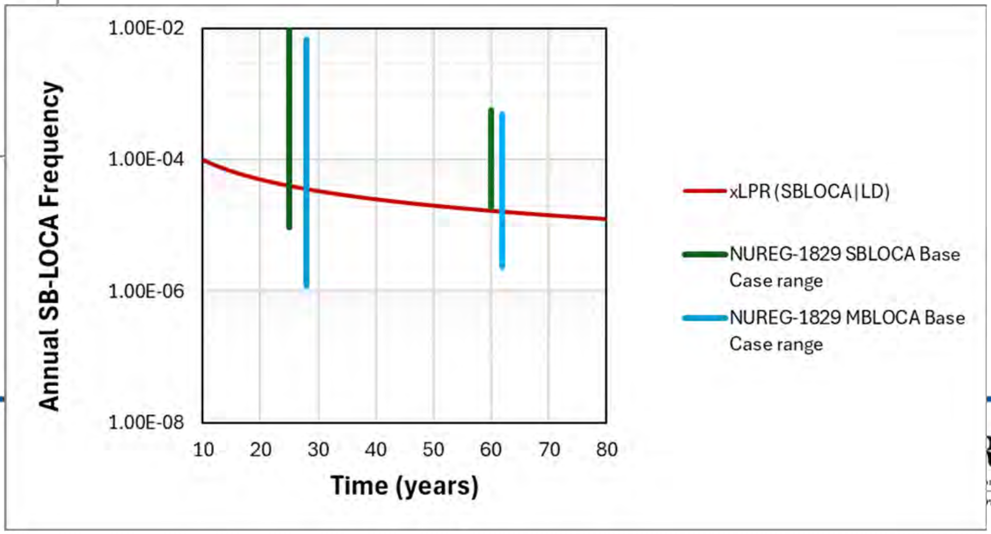


The BWR probabilistic fracture mechanics analyses were also within the range of the NUREG-1829 base case results.

BWR-1: Recirculation System (28-inch)



BWR-1: Recirculation System (12-inch)



International Operational Database Study

- **Motivation**
  - Participants in NUREG-1829 elicitation based their estimates in part from knowledge from operating experience
  - The basis has changed 20 years later
    - More OE knowledge in the later post 25-year lifetime of reactors
    - New mitigation technologies
- **Objectives**
  - Re-evaluate NUREG-1829 LOCA Frequency estimates with knowledge from post-2004 operating experience

Scope of OpE Review	Plant Type	NUREG-1829		Expert Elicitation 2024		ΔEFPY (2024 vs 2004)
		2004		2024		
		ROY	EFPY	ROY	EFPY	
Domestic Plant	BWR	987.8	839.6	1345.9	1144.0	304.4
	PWR	1615.4	1373.1	2735.4	2325.1	952.0

Analysis Procedure

(1) Calculate piping failure precursor frequencies from OE[1]

   (1) Up to 2004 (l_1)

   (2) Up to 2024 (l_2)

(2) Calculate conditional probability of failure distribution

   (1) Extract CFP from NUREG-1829 LOCA frequency estimates using 2004 precursor failure frequency l_1

(3) Calculate updated LOCA frequencies

   (1) Find new LOCA Frequency uncertainty distribution estimates using found CFP and 2024 precursor failure frequency l_2

$$\rho = l \times CFP$$

ρ= LOCA Frequency Distribution

l= Precursor Failure Frequency – (# failures/(Component x Year))

CFP= Conditional Probability of Failure

1. OE database developed under the Nuclear Energy Agency's Component Operational Experience, Degradation and Ageing Programme (CODAP) provided the source for the OE data.

The operating experience analysis results indicate at least an order of magnitude less than the NUREG-1829 results.

LOCA Category	Effective Break Size (inch)	Plant-Level LOCA Frequency (1/Year) – Statistical Mean Values			
		BWR - Piping		PWR - Piping	
		NUREG-1829	2024 Update	NUREG-1829	2024 Update
4	≥ 7	5.9E-06	2.4E-08	7.6E-07	6.0E-08
5	≥ 18	1.0E-06	4.3E-09	1.3E-07	2.6E-08
6	≥ 41	--	--	1.2E-08	4.0E-10

Improved mitigation technologies such as weld overlays attributed with the reduction

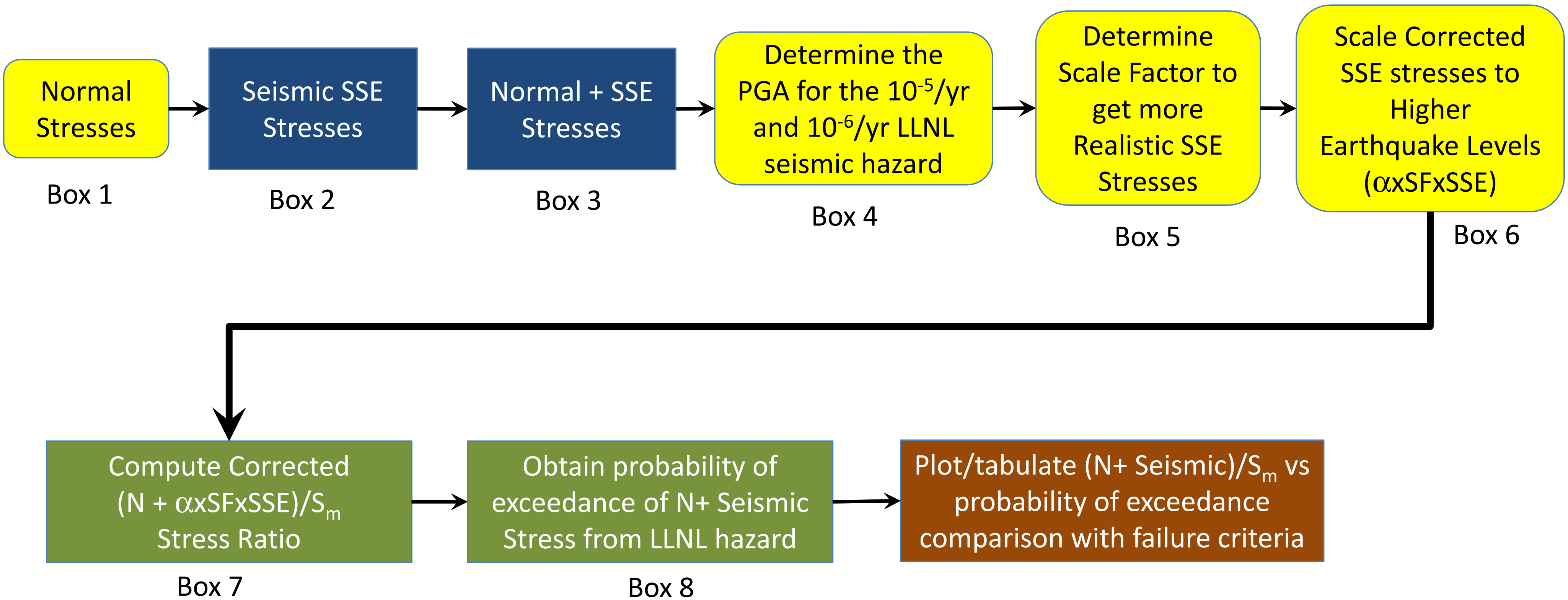
NUREG-1903 Confirmation

- NUREG-1903(ML080880140) addressed potential seismic effects on TBS.
- Evaluated three cases: unflawed and flawed piping failure and indirect piping failure by other components and component supports.
- Used LLNL seismic hazard curves for the assessment.
- For direct unflawed piping, failure probabilities were significantly low compared to the 1E-05 per year frequency used as a basis to establish the TBS.

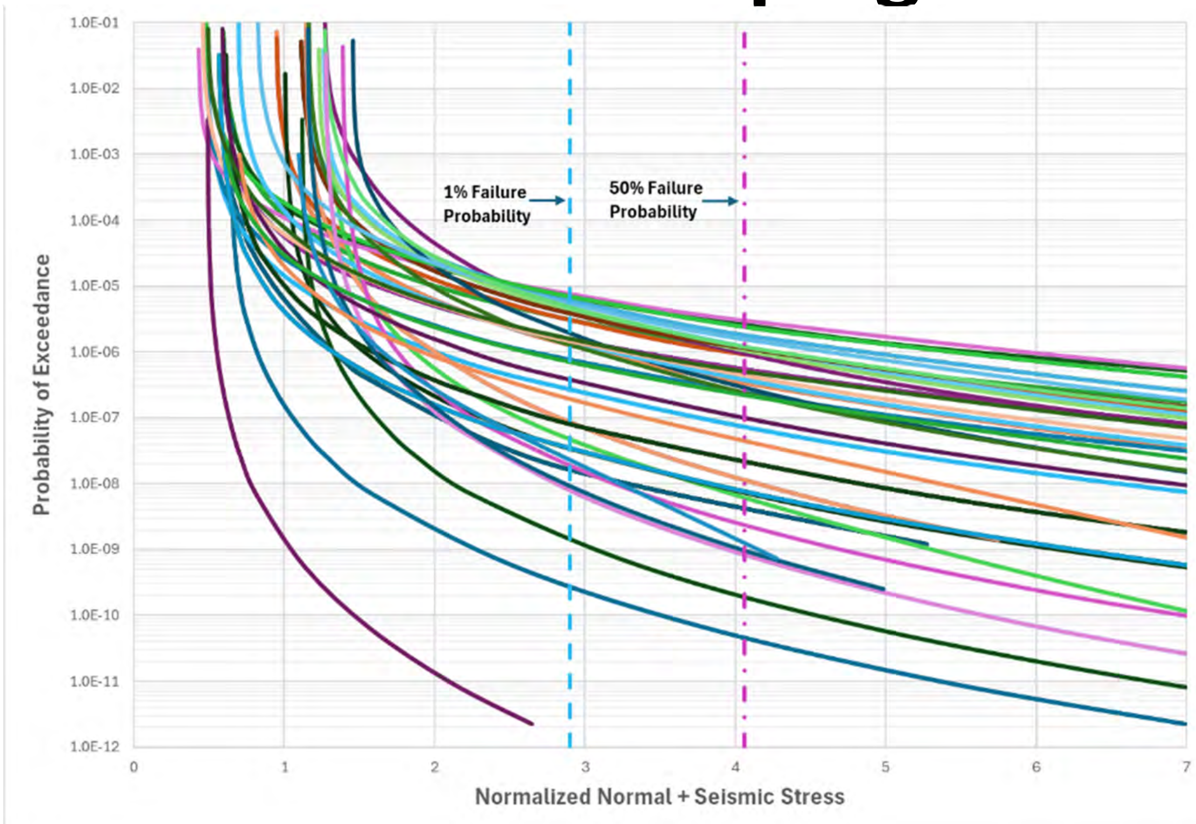
NUREG-1903 Confirmation

- For direct flawed piping, the critical flaws associated with the stresses induced by the 1E-05 and 1E-06 probability of exceedance events were generally large, and the probabilities of pipe breaks larger than the TBS were determined to be less than the TBS frequency criterion.
- For indirect piping failure, the mean probability of failure of the lowest capacity component support was less than 1E-5 for the CE and Westinghouse plants.
- All central and eastern US NPPs recently re-evaluated their seismic hazards (NUREG KM-0017).
- The original assessment results have been updated by using the latest site hazard information for each site (ML24323A205).

Direct Unflawed Piping Failure



Direct Unflawed Piping Failure



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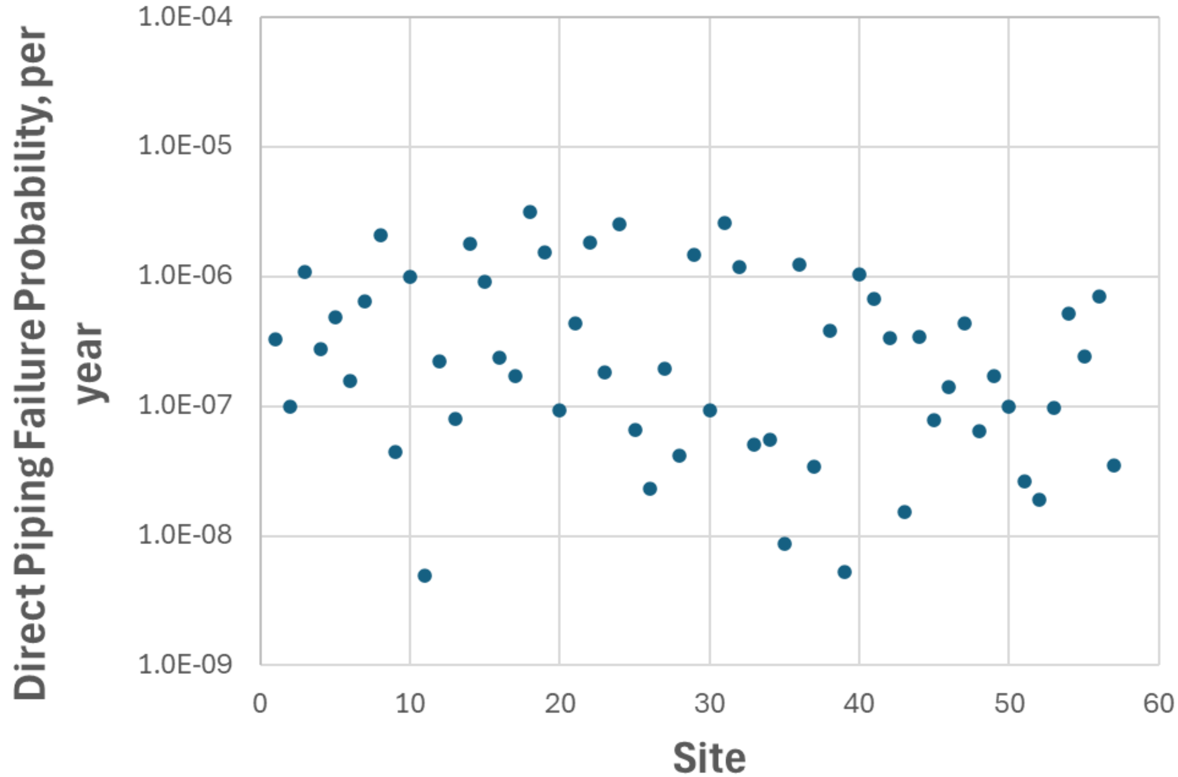
United States Nuclear Regulatory Commission
Protecting People and the Environment

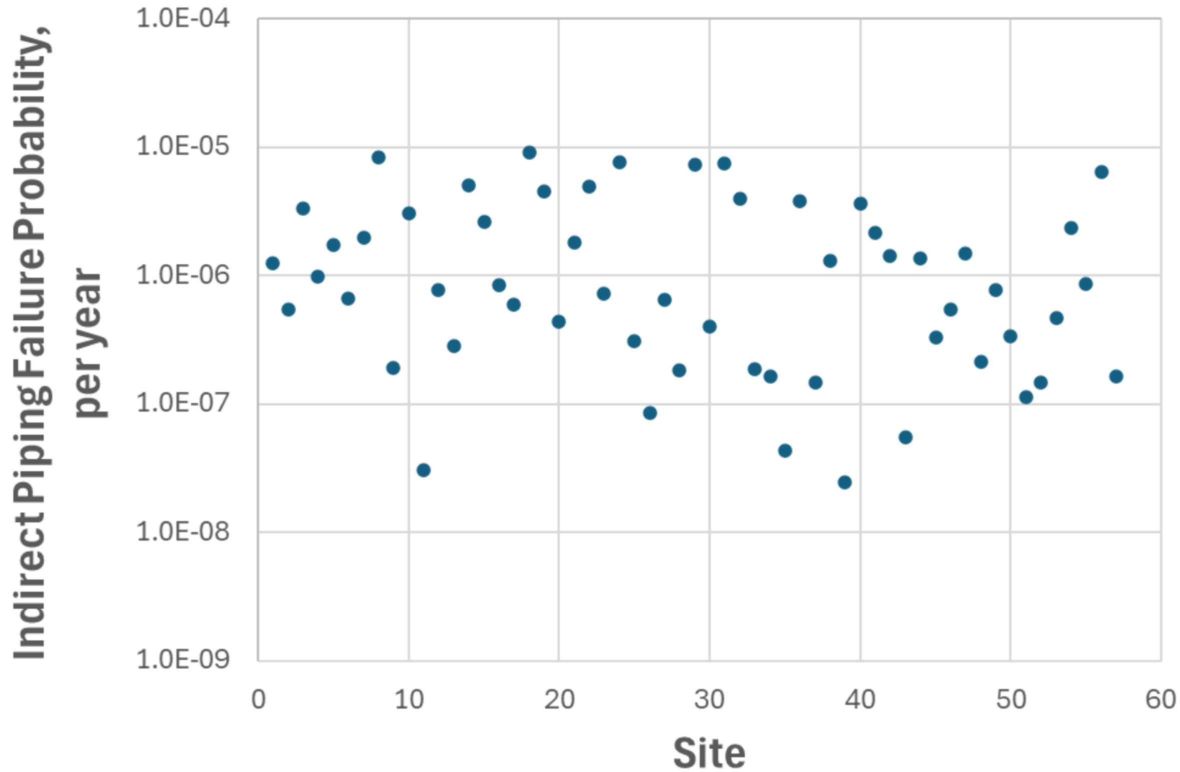
Direct Unflawed Piping Failure Probability



Indirect Piping Failure Probability

- Indirect failures are pipe ruptures caused by failures of major components (e.g., reactor pressure vessel, steam generators, and reactor coolant pumps (RCPs)) or component supports as a result of an earthquake.
- NUREG-1903 indirect piping failure fragility curve
- The results show that mean probabilities of indirect piping failure are all below the TBS frequency criterion of 1E-05.

Indirect Piping Failure Probability



Summary from TBS Confirmation

- LOCA frequencies and TBS are applicable if plant specific applicability is demonstrated.
  – NUREG-1829 and NUREG-1903
  – New designs can develop plant specific TBS
- Inspection of the piping welds with diameters greater than the TBS are needed to ensure LOCA frequencies remain applicable
  – Plants can leverage existing ISI program as needed

Draft Proposed Rule Language for 10 CFR 50.46a

Joseph Messina
Reactor Systems Engineer
Nuclear Methods and Fuel Analysis

Draft Proposed 50.46a Rule Structure

a) Definitions
b) Applicability and scope
c) Application
d) Programmatic requirements
e) ECCS performance
f) Fuel performance criteria
g) Use of NRC-approved fuel in reactor
h) Changes to facility, Technical Specifications, or procedures
i) Authority to impose restrictions on operation
j) Reporting
k) Significant change or error in the ECCS evaluation model

50.46a(a): Definitions

Highlighted definitions:

- Changes enabled by this section: means changes to the facility, Technical Specifications, and procedures that satisfy the alternative ECCS analysis requirements under this section but do not satisfy the ECCS requirements under § 50.46.
- Entity: means an applicant for or a holder of a construction permit, operating license, combined license, standard design approval, or manufacturing license, or an applicant for a standard design certification rule (including such applicant after NRC issuance of a final standard design certification rule).
- Loss-of-coolant accident: means the hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from **breaks in the reactor coolant pressure boundary** up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. LOCAs involving breaks at or below the transition break size (TBS) are design basis accidents. **LOCAs involving breaks larger than the TBS are beyond design basis accidents**.
- Transition break size: for reactors licensed under this part before December 31, 2015, is a break area equal to the largest cross sectional flow area of the reactor coolant pressure boundary piping excluding the hot leg, cold leg, or crossover leg piping for a pressurized water reactor, or the largest cross sectional flow area of either the feedwater line or residual heat removal line inside containment for a boiling water reactor. For reactors that are or will be licensed under this part after December 31, 2015, and for light-water reactors (LWRs) that are or will be licensed under part 52 of this chapter, the TBS will be determined on a plant-specific basis.

50.46a(b) Applicability

- OLs issued prior to Dec. 31, 2015
- Entities whose reactor design is demonstrated under 50.46a(c)(2) to be similar to designs of reactors licensed under part 50 before Dec. 31, 2015

- NUREG-1829 and the TBS based on plants licensed before Dec. 31, 2015. LWRs licensed after may have different piping materials, configurations, and operational and service conditions, among other factors, that may impact the piping break frequencies and thus the TBS
- **Paragraph 50.46a(c)(2)** states that for plants licensed after Dec. 31, 2015, an analysis should be submitted that demonstrates why the proposed reactor design is similar to the designs of reactors licensed under part 50 before Dec. 31, 2015, such that the provisions of this 50.46a may properly apply
  - The analysis must include a recommendation for an appropriate TBS and a justification that the recommended TBS is consistent with the technical basis of this section

50.46(b): Applicability (cont'd): Inspections

Paragraph (b)(3): A licensee must inspect, under § 50.55a(g), for those reactor coolant pressure boundary piping whose inner diameter is greater than the TBS, a sampling of at least 10% of the similar metal piping circumferential welds in a PWR and the circumferential welds in a BWR that are classified as Category A welds before implementation of this section and in every subsequent in-service inspection interval (as defined in § 50.55a(y)). The sampling must include those circumferential welds with the highest failure potential. Credit may be taken for welds inspected as part of established inspection programs (e.g., risk-informed inservice inspection programs). The effect on the TBS of any degradation identified during these inspections must be evaluated.

- This requirement is new from the 2010 draft final rule, added in response to public comments, to reduce the burden of demonstrating plant-specific applicability of the TBS, while providing assurance of safety. Allows licensee to leverage ongoing inspection programs to reduce burden.
- Verifies that analyses that predict component failure remain accurate through the time the component is analyzed, and provides a method to identify novel degradation that may impact the analysis and the structural integrity of the component

50.46a(c): Application

Entities electing to adopt 50.46a will need to provide:

- Evaluation of the applicability of the TBS to the facility
- Weld inspection report
- Description of the risk informed evaluation process for changes made under this rule to meet the risk acceptance criteria in paragraph (h)
- Description of the approach, methods, and decision-making process to be used to evaluate the continued applicability of the TBS
- Description of the non-safety systems credited for LOCAs > TBS and they must be placed in Tech Specs
- Evaluation of leak detection program
- For reactors licensed after 12/31/15, an analysis demonstrating that the reactor design is similar to those licensed before 12/31/15 and the appropriate TBS

50.46a(d): Programmatic Requirements

1) ECCS models and analysis methods are maintained per requirements in (e)(1), (2), and (3)
2) Leak detection systems must be available and used to identify, monitor, and quantify leakage
3) Changes made must be evaluated in a risk-informed evaluation
4) ~~Risk assessments must be maintained and upgraded risk assessments at least once every 5 years~~ – Removed from draft proposed rule during concurrence
5) The effect of all planned facility changes must be evaluated and any changes that would invalidate the evaluation demonstrating the applicability of the TBS cannot be implemented
6) During operation, licensees must perform the (b)(3) weld inspections every subsequent inservice inspection interval (as defined in § 50.55a(y)) on the same samples inspected to satisfy paragraph (b)(3) of this section and evaluate any additional degradation

50.46a(e): ECCS Performance

50.46a(e)(1) establishes two principle ECCS acceptance criteria:

- The ECCS must provide sufficient coolant so that the fuel remains in a coolable geometry during and following the LOCA heatup and quench.
- The ECCS must provide sufficient coolant so that decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the fuel.

❑ Maintaining coolability and removing decay heat has been fundamental to LOCA analysis since the origin of 50.46

50.46a(e)(2): ECCS Performance at or below the TBS

- A number of LOCAs must be analyzed such that there is assurance that the most severe LOCAs at or below the TBS are analyzed
- Uncertainty must be accounted for such that there is a high probability that the ECCS and fuel system acceptance criteria are met
- Changes in fuel geometry must be addressed

❑ Analysis requirements for LOCAs at or below the TBS are essentially unchanged and (still require high probability modeling)

50.46a(e)(3): ECCS Performance above the TBS

- A number of LOCAs must be analyzed such that there is assurance that the most severe LOCAs larger than the TBS up to the double-ended guillotine rupture of the largest pipe in the RCS are analyzed
- There must be assurance to at least a **best-estimate** that the ECCS and fuel system acceptance criteria are met
- Changes in fuel geometry must be addressed
- Calculations may take credit for availability of offsite power
- Do not require assumption of a single failure
- Non-safety-related equipment may be credited if supported by plant-specific data or analysis, and provided that onsite power can be readily provided through simple manual actions to equipment that is credited in the analysis.

❑ **LOCAs > TBS would be beyond-design-basis accidents and be analyzed with best-estimate (best-estimate would refer to nominal and unbiased analyses) modeling, as other beyond-design-basis accidents are (e.g., ATWS and SBO)**

50.46a(f): Fuel Performance Criteria

Fuel system designs must have NRC-approved limits that:

i. Address cladding degradation phenomena
ii. Maintain fuel coolability
iii. Avoid explosive concentration of combustible gas
iv. Demonstrate that, after any calculated successful initial operation of the ECCS, the ECCS must provide sufficient coolant to remove decay heat and prevent further cladding failure for the extended period of time required by the long-lived radioactivity remaining in the fuel.

- Thermal effects of crud and oxide layers must be accounted for

- ❑ Fuel-technology neutral requirements
- ❑ Specific criteria for traditional Zr-UO2 fuel is provided in DG-1263, which would state how the SECY-16-0033 embrittlement research findings should be addressed

Fuel Coolability

While the wording is not significantly different in regards to coolability than 50.46, the NRC staff added a discussion in the FRN Preamble (formerly known as Statements of Consideration) that adds clarification on the interpretation of coolability

- The NRC can envision that some amount of dispersed fuel can remain coolable and safe during a LOCA, therefore the NRC finds that if it can be shown to be safe, then it may be acceptable for LOCAs greater than the TBS
  - Departure from precedent
- The NRC outlined 2 scenarios that remain undesirable though:
  - Widespread brittle failure
  - Fuel or cladding melt
- DG-1434 provides guidance for analyzing the consequences of fuel dispersal

50.46a(g): Use of NRC-Approved Fuel

1) *Fuel load*. A licensee that is approved to use this section may not load fuel into a reactor unless the resulting core design satisfies the ECCS performance requirements of paragraph (e) of this section and the fuel system acceptance criteria and modeling requirements in paragraph (f) of this section, or otherwise complies with Technical Specifications governing lead test assemblies in its license.
2) *Operation*. If a licensee that is approved to use this section determines that fuel in the reactor no longer complies with the ECCS performance requirements of paragraph (e) and the fuel system acceptance criteria and modeling requirements in paragraph (f) of this section, then the licensee must take immediate action to come into compliance with paragraph (e) or (f) of this section, as applicable.

- Clarifies requirement on use of NRC approved fuel designs for which specific ECCS performance requirements have been established.
- Recognizes importance of LTAs for collecting irradiated data to approve new fuel designs.

50.46a(h): Changes Enabled by 50.46a

- Paragraph (h)(1): Changes enabled by this section (50.46a) without prior NRC approval
- Paragraph (h)(2): Changes enabled by this section not permitted under (h)(1)
- Paragraph (h)(3): Criteria that all changes enabled by this section under this section must meet
- Paragraph (h)(4): PRA requirements
- Paragraph (h)(5): Non-PRA requirements

50.46(h)(1): Changes without Prior NRC Approval

Changes enabled by 50.46a are allowed without prior NRC approval if:

i. Change is permitted under 50.59

ii. The NRC-approved risk-informed evaluation process demonstrates that any increases in estimated risk are minimal and the requirements in (h)(3) are met

iii. There is no significant increase in LOCA frequencies and the evaluation demonstrating or establishing the TBS is not invalidated

50.46a(h)(2): Changes Not Permitted Under (h)(1)

For changes enabled by this section not permitted under (h)(1), entities need to submit:

- Information required under 50.90
- Demonstration from the risk-informed evaluation process that the total increases in CDF and LERF are *very small** and the overall risk remains *small*
- Demonstration that the requirements in (h)(3) are met
- Risk-informed evaluation of the cumulative effect on risk on the plant change and all previous changes made under this section
- Demonstration that the ECCS performance criteria are met
- Demonstration that is no significant increase in LOCA frequencies and the evaluation demonstrating or establishing the TBS is not invalidated

*In SRM-SECY-07-0082, the Commission directed that the staff, in the 10 CFR 50.46a draft final rule, should restrict changes to a plant to very small risk increases. Very small risk increase corresponds to an increase in CDF of 1E-6 per reactor year and an increase in LERF of 1E-7 per reactor year.

50.46a(h)(3): Requirements for All Changes Enabled Under 50.46a

All Changes made under this section must meet the following criteria

- Adequate defense-in-depth is maintained
- Adequate safety margins are retained to account for uncertainties
- Adequate performance-measurement programs are implemented to ensure that the risk-informed evaluation continues to reflect actual plant design

50.46a(h)(4): PRA Requirements

PRAs used for the risk-informed evaluation must:

i. Address initiating events, from sources both internal and external to the plant and for all modes of operation, that would affect the regulatory decision in a substantial manner;

ii. Reasonably represent the current configuration and operating practices at the plant;

iii. Have sufficient technical acceptability (including consideration of uncertainty) and level of detail to provide confidence that the total risk estimates and the change in total risk estimates adequately reflect the plant and the effect of the proposed change on risk; and

iv. Be determined, through peer review, to meet industry standards for PRA quality that have been endorsed or otherwise found acceptable by the NRC.

- Expect that low power and shutdown conditions will be addressed with non-PRA methods

50.46a(h)(5): Non-PRA Requirements

Whenever risk assessment methods other than PRAs are used to develop quantitative or qualitative estimates of changes to risk in the risk-informed evaluation, an integrated and systematic process must be used. All aspects of the analyses must reasonably reflect the current plant configuration and operating practices and applicable plant and industry operating experience.

50.46a(i): Authority to impose restrictions on operation

The Director of the Office of Nuclear Reactor Regulation may impose restrictions on reactor operation if the NRC finds that the submitted evaluations of ECCS cooling performance are not consistent with the requirements of this section.

- Maintains the authority of the director of NRR to impose restrictions on operation if there are problems found in a licensee's ECCS evaluation
- This authority has existed since the origin of 50.46 and in the draft final 50.46a and 50.46c rules

50.46a(j): Reporting

(j)(1) and (j)(2): ECCS Reporting and corrective actions

- Eliminates reporting requirements for changes or errors that do not result in an inability to assure compliance with § 50.46a until an SDA or a DC is referenced in an application for a CP, OL, COL, or ML.
  - Parallels what is proposed in the Part 50/52 Alignment rulemaking (ACRS letter: ML22069A269)
- Otherwise, it simply clarifies existing reporting requirements

(j)(3): Risk Assessment reporting

- ~~As part of the risk assessment maintenance and updating required under 50.46a(d)(4), if the re-evaluation results in exceeding the acceptance criteria, must report and explain the changes in PRA modeling, plant, designs, or plant operation that led to the increase(s) in risk no more than 60 days after completing the PRA re-evaluation~~
- Removed from draft proposed rule during concurrence

50.46a(j): Reporting (Cont'd)

Minimal changes reporting

- Must report the changes made under (h)(1) involving minimal changes in risk and a brief summary of the basis for the changes not invalidating the plant's TBS every 24 months

Welding inspection reporting

- Must submit the weld inspection report within 120 days after completing the outage with details of the results of the inspections and the evaluation of the effects on the TBS of any additional degradation since the previous evaluation.
- Can be combined with the summary report required under 50.55a(b)(2)(xxxii)

50.46a(k): Significant changes/Error in ECCS Evaluation Models

For LOCAs at or below TBS, a significant change for UO2 or MOX fuel within cylindrical zirconium-alloy cladding:

i. A calculated peak fuel cladding temperature different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable evaluation model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F; or

ii. A calculated integral time-at-temperature different by more than 1.0 percent equivalent cladding reacted from the oxidation calculated for the limiting transient using the last acceptable evaluation model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective oxidation changes is greater than 1.0 percent equivalent cladding reacted.

- Maintains threshold for significant change in calculated PCT at 50 °F
- Adds a new threshold for significant change in integral time-at-temperature of 1.0% ECR
  - Matches what was in draft final 50.46c rule

50.46a(k): Significant changes/Error in ECCS Evaluation Models

2) **<u>For LOCAs above the TBS</u>**, a significant change or error in the ECCS evaluation model for uranium oxide and mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding is one that results in a significant reduction in the capability to meet the requirements of paragraphs (e)(1) and (f) of this section.
3) For fuel that does not consist of uranium or mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding, a significant change in the ECCS evaluation model is one that results in a significant reduction in the capability to meet the requirements of paragraphs (e)(1) and (f)(1) of this section.

- For breaks above the TBS, an entity could define alternative criteria for a significant change. If alternative criteria are not defined, then the same reporting criteria in proposed 10 CFR 50.46(k)(1) would be applied (50°F PCT and 1% ECR)
- A new definition of significant change or error may be necessary for other fuel/cladding materials

Changes to Base 50.46

- Corresponding changes were made to 50.46
  - For example, that either 50.46 or 50.46a could be used along with applicability statements
- The 50.46(b) criteria (e.g., 2200°F PCT and 17% ECR limits) were *not* changed for licensees who do not elect to adopt 50.46 as an attempt to limit the scope of the rule and possible delays
- While the applicability of 50.46 criteria was not expanded from UO2 pellets within Zircaloy or ZIRLO cladding, a statement was added to say that the criteria in 50.46(b) or 50.46a(f)(1) must be met
  - 50.46(3)(i): "The ECCS system must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents (LOCAs) conforms to the criteria set forth in paragraph (b) of this section *or § 50.46a(f)(1)*..."
  - **Therefore entities with that do not adopt 50.46a and have non-Zircaloy or non-ZIRLO fuel can either elect to submit an exemption to 50.46 to use the 50.46(b) criteria or elect to use the fuel-technology-neutral criteria in 50.46a(f)(1) without an exemption**
    - Entities that elect to use the 50.46a(f)(1) criteria would be expected to address the SECY-16-0033 embrittlement research findings

Questions

U.S.NRC
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Protecting People and the Environment

Draft Guide-1426
An Approach for a Risk-Informed Evaluation Process Supporting Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Reactors

Kristy Bucholtz
Reliability and Risk Analyst
PRA Oversight Branch

50.46a Risk Approach

- 50.46a(c)(iii) requires a risk-informed evaluation to make any proposed change.
- 50.46a has two risk pathways:

  (1) Individual submittals - risk-informed evaluation with a risk assessment of the proposed changes.

  - Multiple submittals are allowed
    - Initial adoption of the rule
    - Each future change enabled by 50.46a.
  - Initial and each enabled change submitted to NRC for review.
  - Paragraph (h)(2) applies → Acceptance guidelines in DG-1426, Section 2.2.3.1 apply.

50.46a Risk Approach

- 50.46a has two risk pathways:

  (2) Risk-Informed Evaluation Process (RIEP) submittals - risk-informed evaluation with risk assessment of the proposed changes, **however, RIEP is also submitted for NRC approval**.

  - Multiple submittals are allowed
    - Initial adoption of the rule which includes the RIEP.
    - Each future change enabled by 50.46a.
  - Each future change enabled by 50.46a is evaluated with the RIEP.
  - Paragraph (h)(1) → if met, the change may be made without NRC approval. → DG-1426, Section 2.2.3.2.
  - Paragraph (h)(2) → if met, submit for NRC approval → DG-1426, Section 2.2.3.1.

DG-1426 Structure

- **DG-1426 follows the same structure as RG 1.174**
  - C.1 - Element 1: Define the proposed change
  - C.2 - Element 2: Perform Engineering Analysis
  - C.3 - Element 3: Define Implementation and Monitoring Program
  - C.4 - Element 4: Submit the License Amendment Request
  - C.5 - Quality Assurance
  - C.6 - Documentation

Element 1

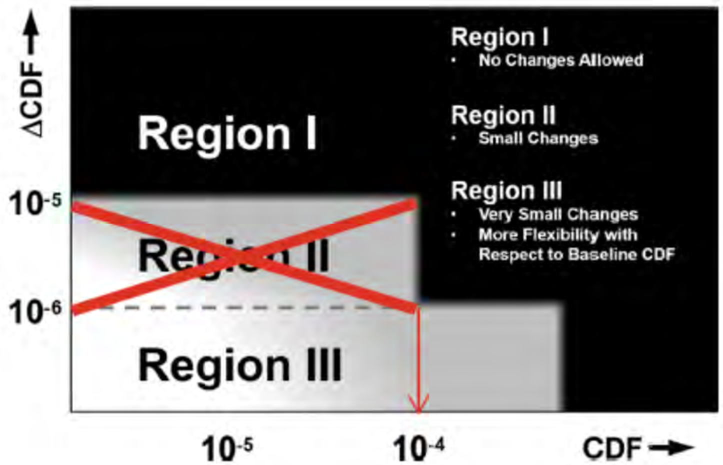
- **C.1 - Element 1: Define the proposed change**
  - DG-1426 follows the structure of RG 1.174, which is written for licensees changing their licensing basis, with a few minor changes, identified below:
    - Modified from “licensee” to “entity.”
    - Modified from “licensing basis changes” to “proposed changes.”
    - Added sentences for entity to identify the aspects that may be affected, such as the licensing basis or entity-controlled documentation, and implementation pathway.
      - NRC review - license amendment via 10 CFR 50.90
      - Without NRC review

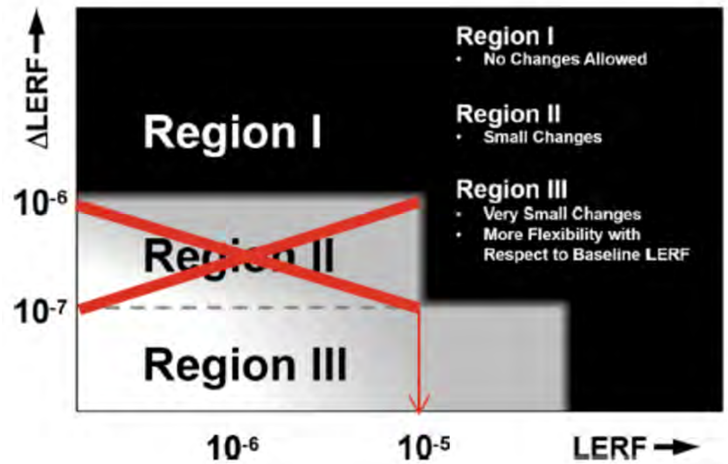
Element 2

- **C.2 - Element 2: Perform Engineering Analysis**
  - Section 2.1 - Risk-Informed Evaluation Process
  - Section 2.2 - Risk Assessment
    - Section 2.2.1 - Probabilistic Risk Assessment (PRA)
      - Addresses scope, level of detail, technical elements, and plant representation.
      - Based on RG 1.200.
    - Section 2.2.2 - Non-PRA risk assessments
    - Section 2.2.3 - Risk Metrics
      - Section 2.2.3.1 - Acceptance guidelines for risk-informed evaluations requiring prior NRC approval
      - Section 2.2.3.2 - Acceptance guidelines for self-approved risk-informed evaluations
    - Section 2.2.4 - Defense in Depth
    - Section 2.2.5 - Safety Margins

Element 2

- Section 2.2.3.1 - Acceptance guidelines for risk-informed evaluations requiring prior NRC approval

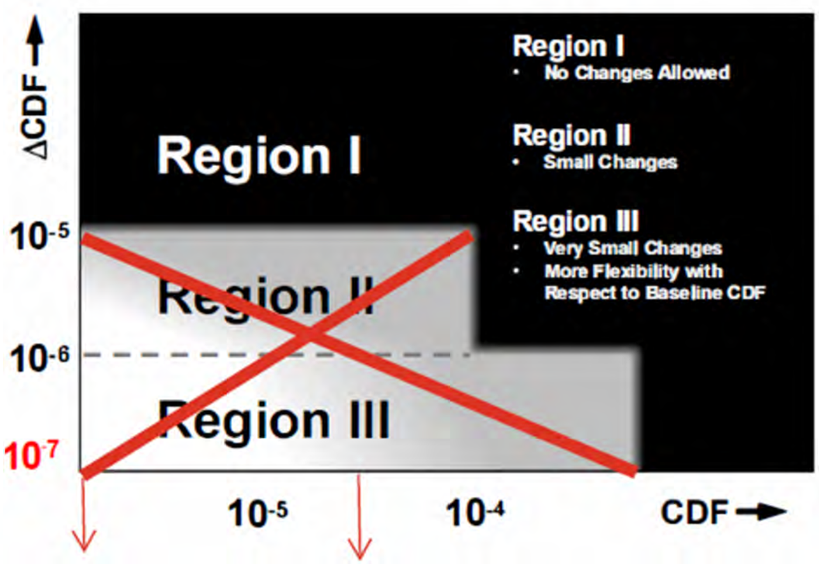


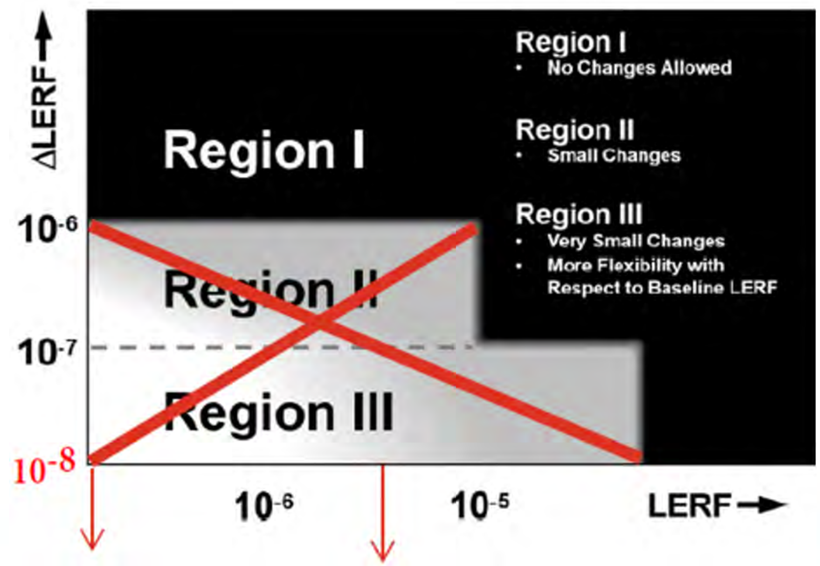


5. In order to more closely follow the approach presented in Regulatory Guide 1.174, the staff should modify the proposed rule to ensure that any changes under this rule be further restricted to very small risk increases, notwithstanding the fact that they would otherwise be permitted under 50.59. Therefore, staff should add the word "very" before the word "small" in section (f)(1)(i) so that it reads "...the total increase in core damage frequency and large early release frequency are very small and the overall risk remains small..." or make other changes as appropriate to achieve the above objective.

Element 2

- Section 2.2.3.2 - Acceptance guidelines for self-approved risk-informed evaluations





Element 3

- **C.3 - Element 3: Define Implementation and Monitoring Program**
  - Pointer to the section in RG 1.174 for implementation and monitoring program
    - The periodic updating of the risk assessments was removed from the draft proposed 50.46a rule and DG-1426.
    - DG-1426 no longer includes re-evaluation of the risk assessment.
    - Staff removed the requirement to update the PRA every 5 years from the draft proposed 50.46a rule.
    - The update remains in DG-1426, but has been changed from a "must" to a "should."

*Note: Staff is revising the version of DG-1426 that was originally submitted to the ACRS to address the four changes listed above.

Element 4

- **C.4 - Element 4: Submit the License Amendment Request**
  - Submit a summary of the PRA model and methods used to evaluate the proposed change.
    - which risk methods are used and why they are acceptable,
    - key modeling assumptions and consideration of uncertainty,
    - key operator actions, and
    - changes required to event or fault trees in the PRA model.
  - For RIEP submittals, submit details of the RIEP to be used to support changes without NRC approval.
    - Description of the entity's PRA model and any non-PRA risk assessment methods to be used.
    - Description of the entity's approach, methods, and decisionmaking process to evaluate:
      - Risk criteria
      - Defense in depth
      - Safety margins
      - Performance measurement and monitoring

Quality Assurance and Documentation

- **C.5 - Quality assurance**
  - The same as RG 1.174, with no substantial changes.

- **C.6 - Documentation**
  - Differs from RG 1.174
    - For each plant change, the entity should document the risk-informed evaluation, consistent with section C.4 of RG 1.200.

Questions

U.S.NRC
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Proposed Rule: Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors

December 17-18, 2024

Opening Remarks

Theresa Clark
Director
Division of Safety Systems

Draft Regulatory Guides for Zirconium-Alloy Cladding Analytical Limits

James Corson
Senior Reactor Systems Engineer (Fuels Analyst)
Fuel and Source Term Code Development Branch

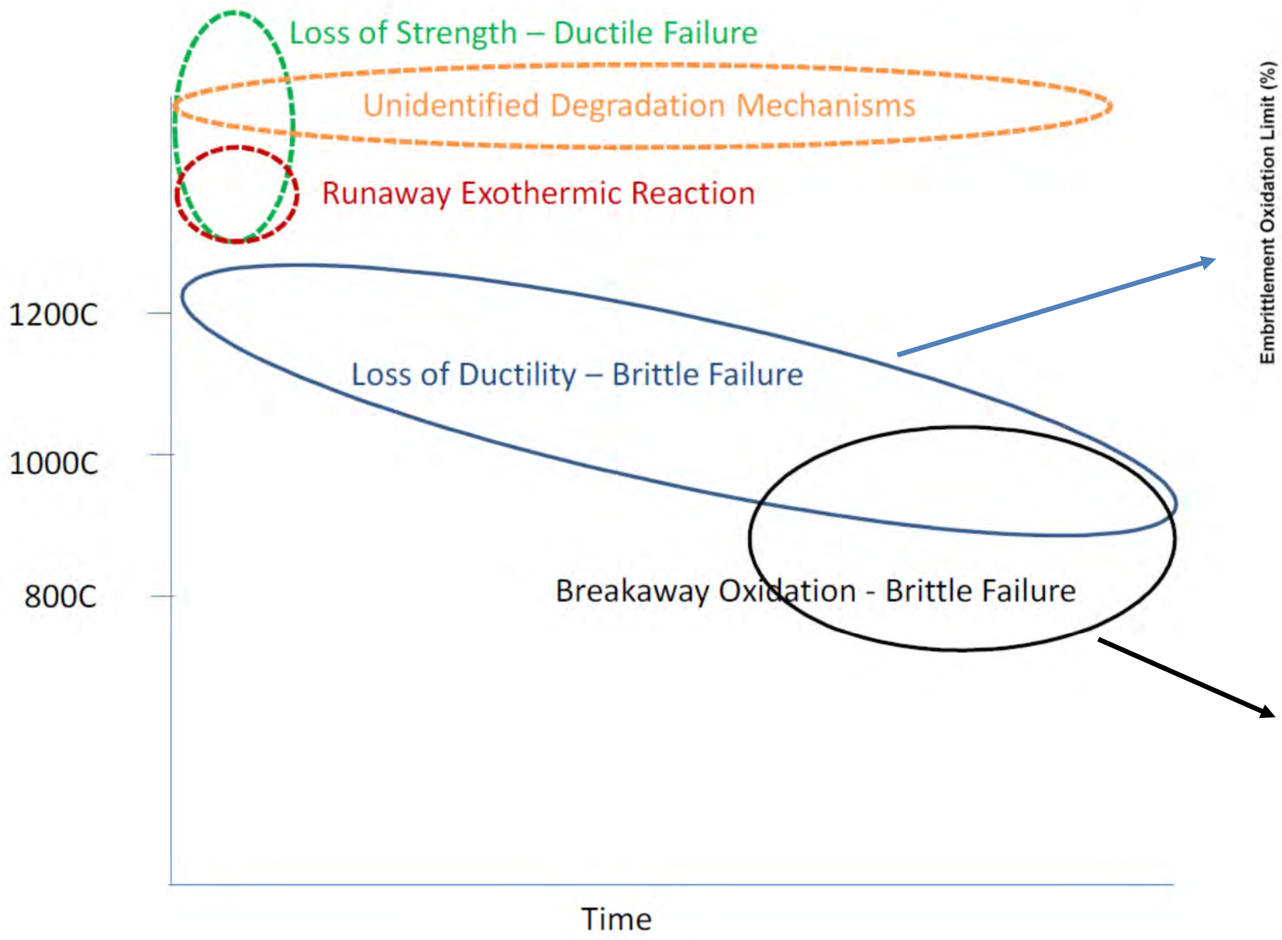
Overview

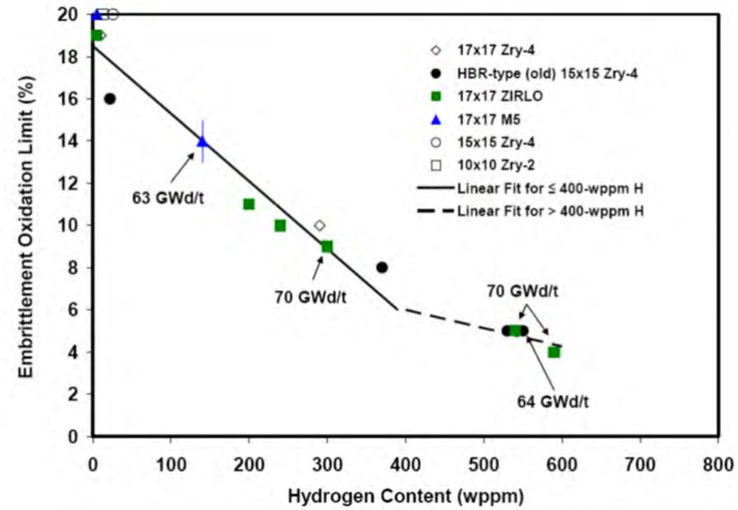
- As part of the 50.46c rulemaking, NRC staff developed 3 draft regulatory guides to address zirconium-alloy cladding analytical limits
  - DG-1261 (RG-1.222): Measuring Breakaway Oxidation Behavior
  - DG-1262 (RG-1.223): Determining Post-Quench Ductility
  - DG-1263 (RG-1.224): Establishing Analytical Limits for Zirconium-Based Alloy Cladding
- Staff have updated these documents to reflect the 50.46a proposed rule language
  - Updates reflect fact that aspects of the 50.46c rule language have been moved to guidance for 50.46a proposed rule
  - Otherwise, the guides are (mostly) unchanged from the versions included in the 50.46c draft final rule package

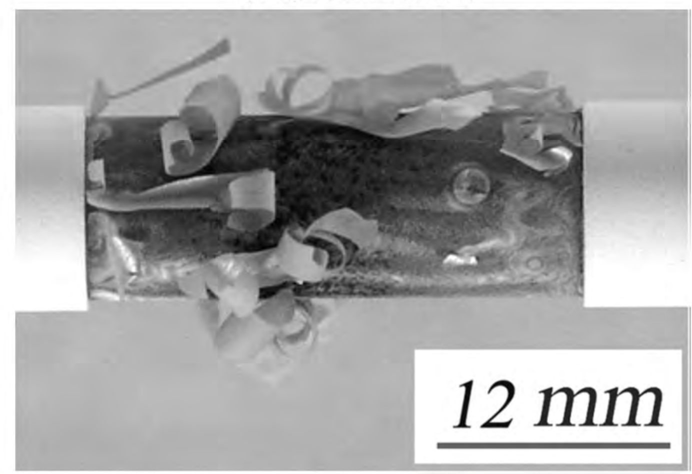
Relation of DGs to the Rule Language

- 50.46a(f)(1) *Fuel performance criteria.* Fuel system designs must have NRC-approved limits that:
  i. Address cladding degradation phenomena;
  ii. Maintain fuel coolability
  iii. Avoid explosive concentration of combustible gas; and
  iv. Demonstrate that, after any calculated successful initial operation of the ECCS, the ECCS must provide sufficient coolant to remove decay heat and prevent further cladding failure for the extended period of time required by the long-lived radioactivity remaining in the fuel.
- DGs 1261, 1262, and 1263 primarily address zirconium-alloy cladding embrittlement
  - Thus, they mostly address 50.46a(f)(1)(i), though DG-1263 includes a limit for 50.46a(f)(1)(iii)

Zirconium Alloy Cladding Degradation Mechanisms







Abbreviated History

- DGs 1261, 1262, and 1263 included in the 50.46c proposed rule published March 2014 (ML12283A174)
  - Public comment period ended August 2014
  - Several public meetings held on 50.46c during public comment period
- Several public meetings held between close of public comment period and publication of draft final rule in 2016
  - Public meeting on regulatory guidance in April 2015 (ML15132A743)
  - Overview of preliminary draft changes to the rule and guidance in October 2015 (ML15321A004)
  - ACRS SC on the draft final rule package on November 3, 2015 (ML15320A187)
  - ACRS FC meeting on the draft final rule package on December 3, 2015 (ML15349A717)
- DGs included as RGs 1.222, 1.223, and 1.224 in draft final rule package published March 2016 (ML15238A933)

DG-1261: Measuring Breakaway Oxidation Behavior

– Breakaway oxidation in zirconium alloy cladding associated with change from protective tetragonal oxide to non-protective monoclinic oxide

– Breakaway oxidation characterized by significant increase in oxidation rate and hydrogen pickup, both of which lead to cladding embrittlement

– NRC's LOCA program showed that minor changes in alloy composition or manufacturing processes can have significant impact on breakaway oxidation behavior

DG-1261

- Defines an experimental technique capable of determining the effect of composition changes or manufacturing changes on the breakaway oxidation behavior
- Experimental technique includes flexibility, where possible, to allow variation of equipment and procedures in use at other laboratories
- Discusses both initial testing and periodic confirmatory testing

DG-1261

- Initial testing includes examination of breakaway oxidation behavior at a range of temperatures to identify the critical temperature associated with the shortest time to breakaway oxidation
- Allows adoption of Argonne National Laboratory test data for initial implementation of 50.46a rule

DG-1261

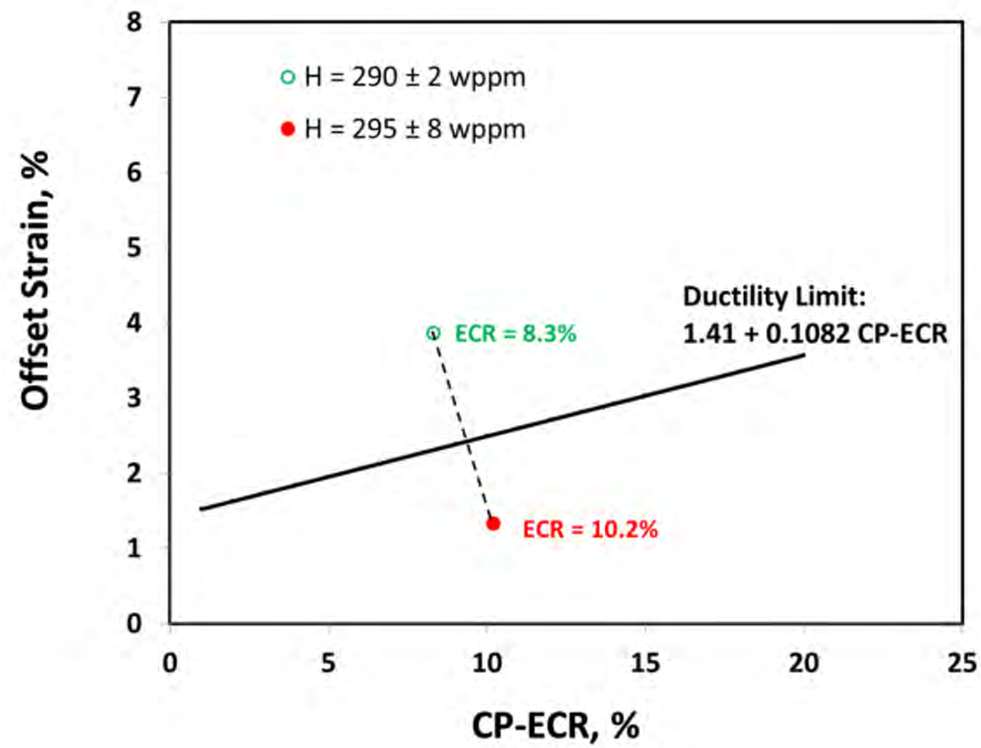
- Periodic testing is used to confirm the initial testing (and associated analytical limit) remains applicable to manufacturing life-cycle
  - Periodic Confirmatory Test Program Plans (PCTPP) would be developed by each cladding vendor and submitted for NRC review and approval
  - Periodic testing is focused only on the critical temperature identified in initial testing
  - Vendors would define periodic testing frequency in the PCTPP; DG-1263 provides an optional default frequency (testing once per ingot) and states that other frequencies could be reviewed and approved
  - Guidance allows for relaxation of test frequency with time
  - Periodic testing results are not submitted but must be available for audit

DG-1262: Determining Post-Quench Ductility

- Defines an experimental technical to measure the ductile-to-brittle transition for the zirconium-alloy cladding material
- Experimental technique includes flexibility, where possible, to allow variation of equipment and procedures in use at other laboratories
- Provides detailed discussion of determining the ductile-to-brittle transition CP-ECR for a given hydrogen level, allows for binning results with similar H content
  - CP-ECR = equivalent cladding reacted calculated using the Cathcart-Pawel correlation

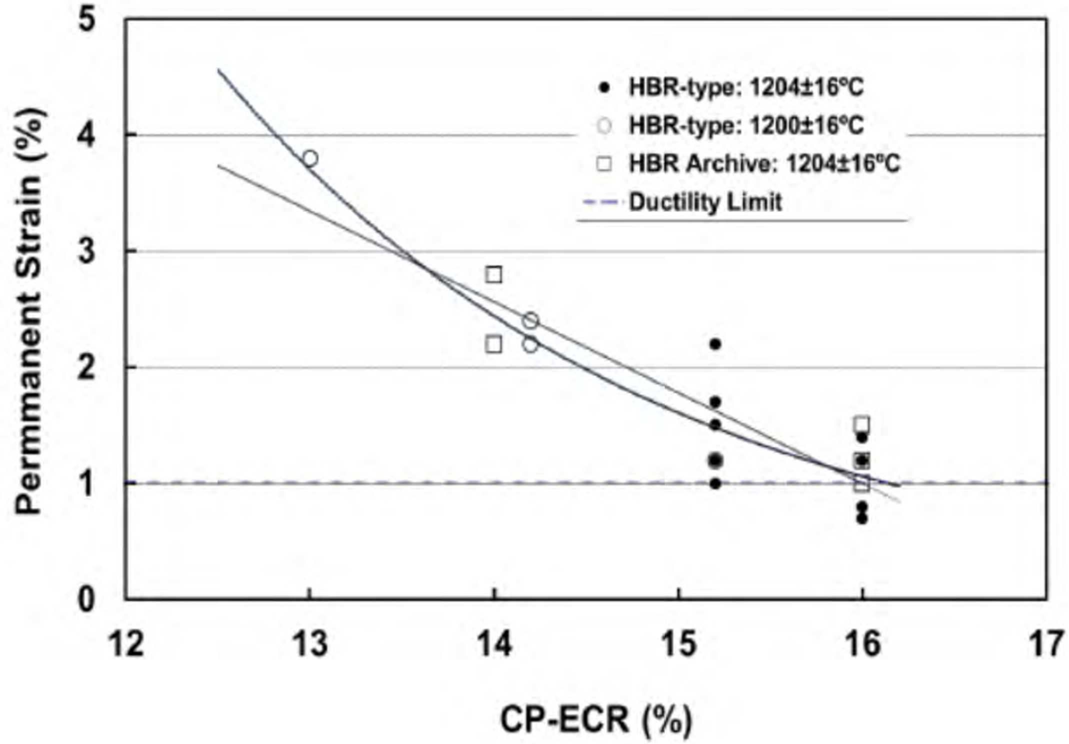
DG-1262

- Two approaches (set and curve-fit) are provided to address expected data scatter in a data “bin” and determine the ductile-to-brittle transition CP-ECR for a given hydrogen level
- Set approach



DG-1262

- Two approaches (set and curve-fit) are provided to address expected data scatter in a data “bin” and determine the ductile-to-brittle transition CP-ECR for a given hydrogen level
- Curve-fit approach



DG-1263: Establishing Analytical Limits for Zirconium-Alloy Cladding Material

- Describes an approach to establish limits to address zirconium-alloy cladding degradation phenomena
  - Analytical limits for post-quench ductility and breakaway oxidation
  - PCT limit to address post-quench ductility also protects against higher-temperature degradation mechanisms
- Provides guidance on how to consider the impact of oxygen diffusion from inside surfaces on cladding degradation
- Provides default cladding hydrogen uptake models for currently approved cladding models
- Provides an analytical limit for combustible gas generation

DG-1263

C.1.A – An acceptable limit for currently deployed alloys

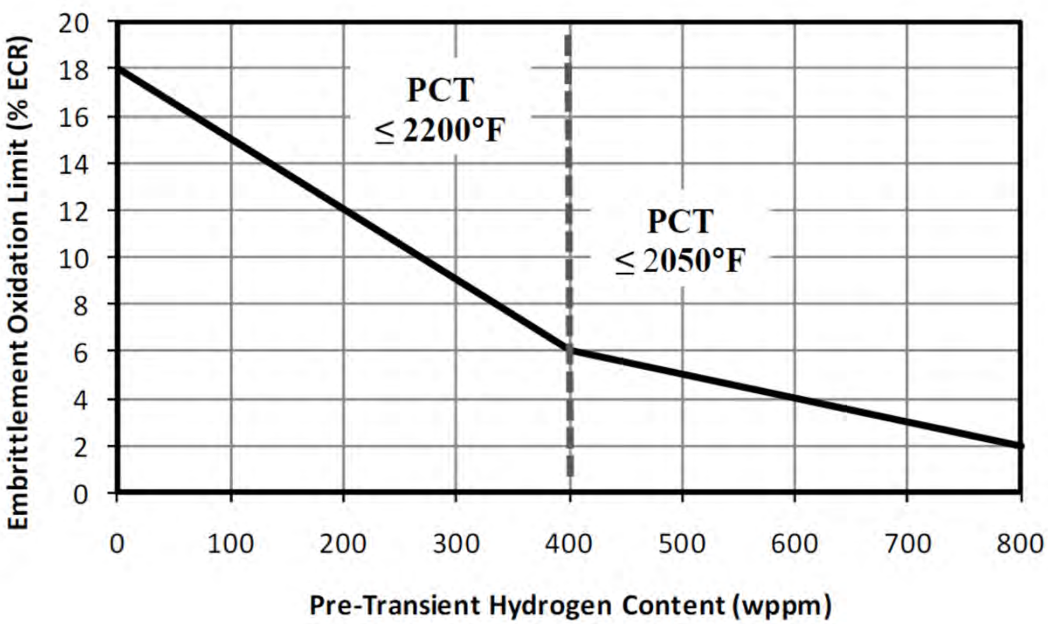


Figure 2 of DG-1263. Acceptable analytical limits for peak cladding temperature and integral time at temperature (as calculated in local oxidation calculations using the CP correlation) for Zircaloy-2, Zircaloy-4, ZIRLO®, M5®, and Optimized ZIRLO™

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DG-1263

C.1.B – Adopting Figure 2 for New Alloys

New alloys can adopt Figure 2 by providing the measured ductile-to-brittle transition level for cladding material in the following conditions:

1. As received
2. Unirradiated, pre-hydrided within 100 ppm of the maximum hydrogen content specified at end of life (EOL)
3. Unirradiated, pre-hydrided within 100 ppm of half of the maximum hydrogen content specified at EOL
4. Irradiated (unless the new alloy is "similar" to previously tested alloys)

DG-1263

- New cladding alloys are considered "similar" to alloys tested in NRC's LOCA program (conducted at Argonne National Laboratory) if they:
  - Use the Kroll process
  - Operate less than or equal to the maximum fluence
  - Include only the alloying elements present in the materials tested
  - Have similar alloying content of each element to the materials tested in NRC's LOCA program, whereby each alloy element is defined by less than or equal to 25 percent deviation from the alloying limits defined for the tested alloy

DG-1263

C.1.C & C.1.D – Adopting other post-quench ductility limits

- Analytical limits other than those defined in Figure 2 can be adopted for new and existing alloys to gain margin for superior alloy-specific cladding performance (C.1.C) or for slower embrittlement behavior at lower temperatures (C.1.D)

DG-1263

C.1.E – Hydrogen pickup models

- An alloy-specific cladding hydrogen uptake model should be used in conjunction with the hydrogen-dependent embrittlement threshold provided in Figure 2
- Appendix A of DG-1263 provides acceptable fuel rod cladding hydrogen uptake models for the current commercial zirconium alloys

DG-1263

C.1.F – Demonstrating compliance for PQD

- Identify the limiting conditions and assumptions that maximize predicted PCT and local oxidation
- Demonstrate PCT and max local oxidation are below PQD analytical limit
- Provides allowance for subdividing the ECCS evaluation based on cladding hydrogen content, burnup, fuel rod power, or a combination
- Provides allowance to use Figure 2 for legacy fuel to show compliance with 50.46a(f)(1) requirements

DG-1263

C.2 – Breakaway oxidation

- Provides allowance for legacy fuel to use analytical limit established for the current version of the alloy
- Applicants may elect to establish the analytical time limit for breakaway oxidation with conservatism relative to the measured minimum time (i.e., reduce the time) to the onset of breakaway oxidation
- The total time that the cladding is predicted to remain above the temperature that the zirconium-alloy cladding material has been shown to be susceptible to breakaway oxidation (800^{o}C default) must be less than the analytical limit
  - Applicant may credit operator action to limit the duration at elevated temperatures provided these actions are consistent with existing procedures and the timing of such actions is validated by operator training on the plant simulator or via a job performance measure

DG-1263

C.3 – Hydrogen generation

- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam should not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react
- Same as existing requirement in 50.46(b)(3)

DG-1263

C.4 – Inner diameter oxidation

- ECCS evaluation models should consider oxygen diffusion from the cladding inside surfaces if an oxygen source is present on the inside surface of the cladding at the onset of the LOCA
  - Cladding rupture: calculate 2-sided oxidation using CP correlation, consider the reduced cladding thickness and the rupture mid-plane an apply Figure 2 (same approach used today)
  - Fuel-cladding bond: calculate 2-sided oxidation once the fuel-cladding bond layer is predicted to occur (default threshold is 30 GWd/MTU, but higher limits can be proposed for NRC review and approval)

Conclusions

- Three draft guides have been developed to support performance-based criteria related to zirconium-alloy cladding degradation in the 50.46a proposed rule. The DGs provide guidance to develop material-specific analytical limits on key embrittlement mechanisms.
  - DG-1263 also includes a hydrogen generation limit
- The DGs are based on the guides submitted as part of the 50.46c draft final rule package but have been updated to reflect requirements of the 50.46a proposed rule (e.g., removing specific limits from the rule language)
  - DGs reflect extensive interactions with industry stakeholders during the 50.46c draft final rule public comment period.

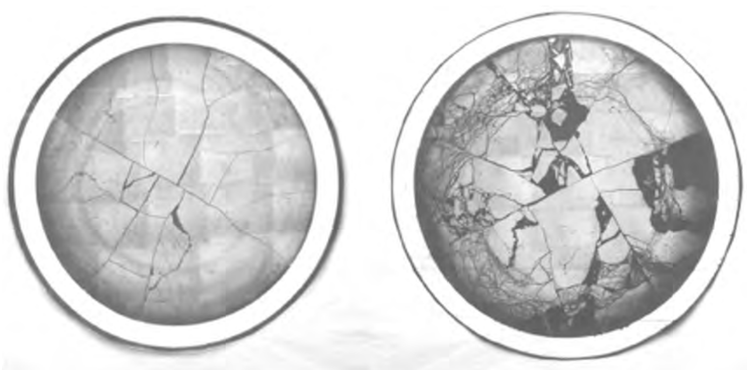
Questions

DG-1434: Addressing the Consequences of Fuel Dispersal in Light-Water Reactor Loss-of-Coolant Accidents

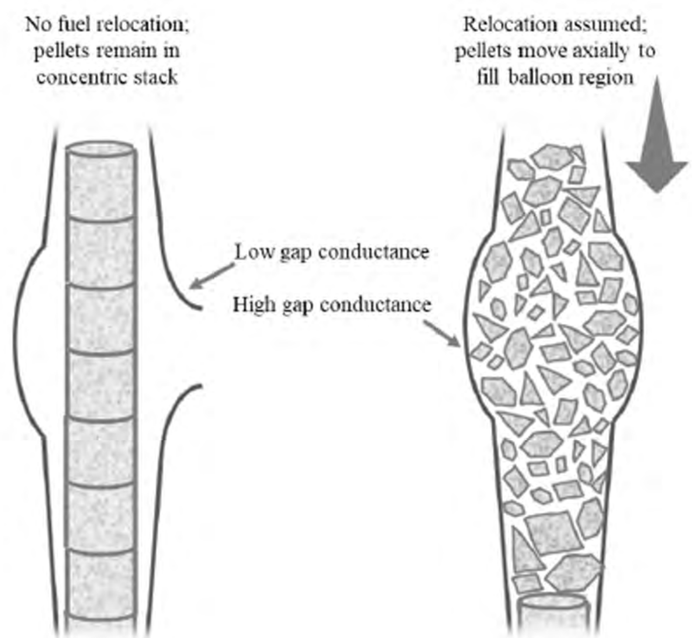
James Corson
Senior Reactor Systems Engineer (Fuels Analyst)
Fuel and Source Term Code Development Branch

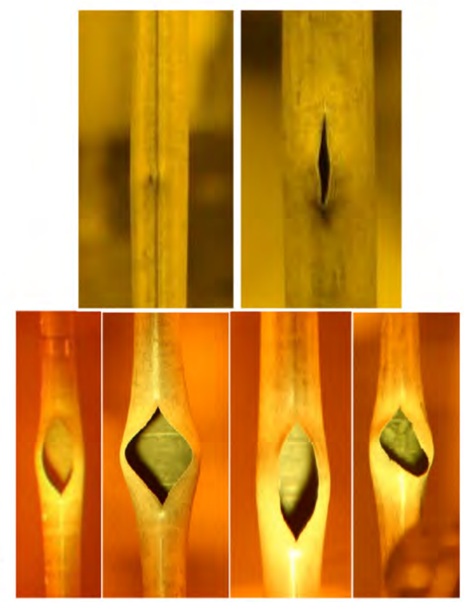
Fuel Fragmentation, Relocation, and Dispersal (FFRD)

- At HBU experiments have shown that the fuel can fragment during a LOCA
- Differences in pressure across the cladding can lead to cladding ballooning and burst
- The fragmented fuel can relocate axially into the balloon region of the fuel rod and if burst occurs, disperse into the RCS



Segment from NRC's ANL LOCA program at 55 GWd/MTU before and after testing





Burst openings from Studsvik LOCA tests (NUREG-2121)

FFRD: History

- 1974 – 10 CFR 50.46 criteria for LOCA’s
- 1980 – FFRD Discovered
- 1984-1995 – GI-92, “Fuel Crumbling During LOCA”
- 2006 – FFRD testing showed gross fuel loss at high BU
- 2008 – RIL-0801
- 2012 – NUREG-2121
- 2015 – SECY-15-0148 – No imminent safety concern
- 2016 – Draft final rule for 50.46c
- 2021 – RIL-2021-13
- 2022 – SRM-SECY-21-0109 – Include FFRD in IE Rulemaking
- 2024 – NUREG/CR-7307

Fuel Dispersal: Background and Regulatory Issue

- The 50.46 acceptance criteria date to 1974 when FFRD were not known phenomena
- Acceptable approaches to demonstrate compliance with the regulations have ensured that catastrophic failure of the fuel rod structure and loss of fuel bundle configuration are precluded
  - Fuel dispersal would be a departure of precedent
- Fuel dispersal is not explicitly addressed within the current regulations
  - Proposed rule language (50.46a) allows for some flexibility regarding fuel dispersal
  - DG-1434 provides guidance for addressing fuel dispersal within the proposed rule

Draft Guidance for Fuel Dispersal

- DG-1434 provides guidance for addressing the impact of fuel dispersal on ECCS performance
  - Includes a model to estimate the mass of dispersed fuel
  - Provides high-level acceptance criteria for fuel dispersal
  - Lists analyses to perform to address consequences of dispersed fuel
- DG-1434 builds on recent research efforts and reflects the current state of knowledge
  - Research Information Letter (RIL) 2021-13, "Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup" (ML21313A145)
  - NUREG/CR-7307, "PIRTs on High Burnup Fuel Fragmentation, Relocation, Dispersal, and Its Consequences" (ML24155A058)
  - EPRI-sponsored 2024 White Paper, "Assessment of Existing Fuel Fragmentation, Relocation, and Dispersal Data: Best Estimate Interpretation"
  - Several recent publications from researchers at Oak Ridge and Idaho National Laboratories

Conclusions from the Fuel Dispersal PIRT

- Understanding how much material disperses is crucial to demonstrating coolability
  - Key parameters influencing dispersal include transient FGR, fuel fragment size distribution, cladding burst characteristics, spacer grid characteristics, core flow patterns during the transient, and core loading pattern
  - Some parameters can be calculated fairly accurately (e.g., core loading pattern, core flow)
  - Other parameters are less well known and highly uncertain (e.g., transient FGR, fragment size distribution, burst opening size, impact of spacer grids on debris trapping)
- Dispersal of fuel fragments remains poorly understood
  - However, the PIRT panelists believe it should be possible to perform simplified analyses to demonstrate coolability so long as the dispersed mass remains low

Structure of DG-1434

1. Limits on applicability
2. FFRD thresholds
3. Analytical limits for fuel dispersal
4. Methods for estimating the dispersed fuel mass
5. Impacts of fuel dispersal
   a. Fuel particle transport and deposition
   b. Fuel coolability and long-term cooling
   c. Re-criticality
   d. Radiological consequences and environmental qualification (covered by RG 1.183 Rev. 2)

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Limits on Applicability

- FFRD thresholds and methods for estimating the mass of dispersed fuel apply to undoped UO_2 fuel in zirconium-alloy cladding
  - Extension to UO_2 fuel with dopants (e.g., gadolinia, chromia, alumina, and/or silica) or MOX will be considered on case-by-case basis
  - Recently completed SCIP-IV tests and upcoming tests under the Second Framework for Irradiation Experiments (FIDES-II) and SCIP-V could help address this limitation
- Other sections of the guidance are generally applicable to all fuel designs, unless otherwise noted
  - For example, some limits related to recriticality (see upcoming slides)

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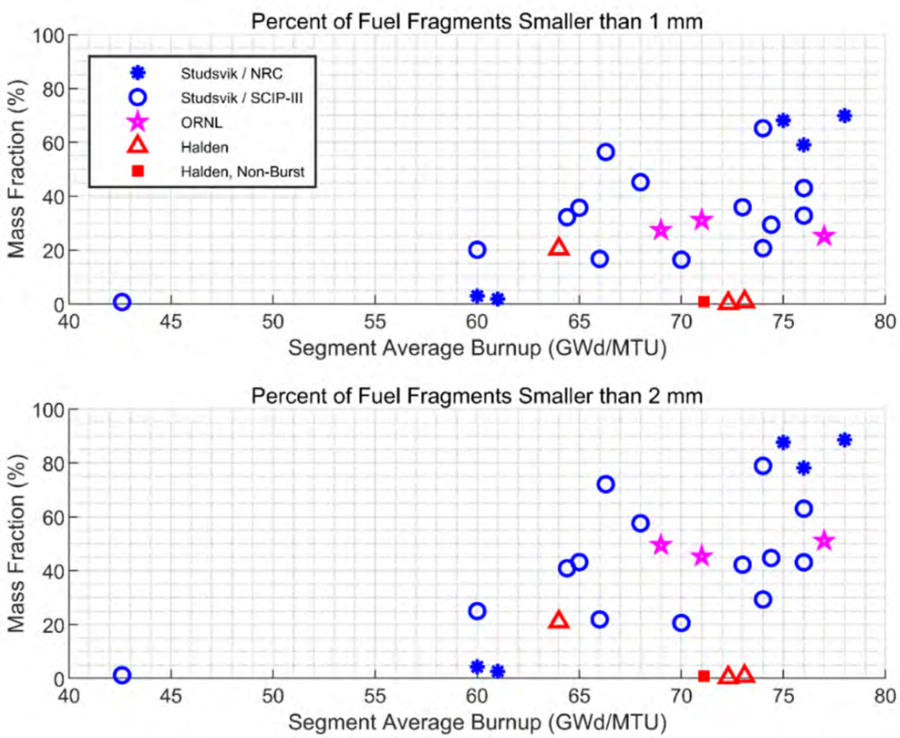
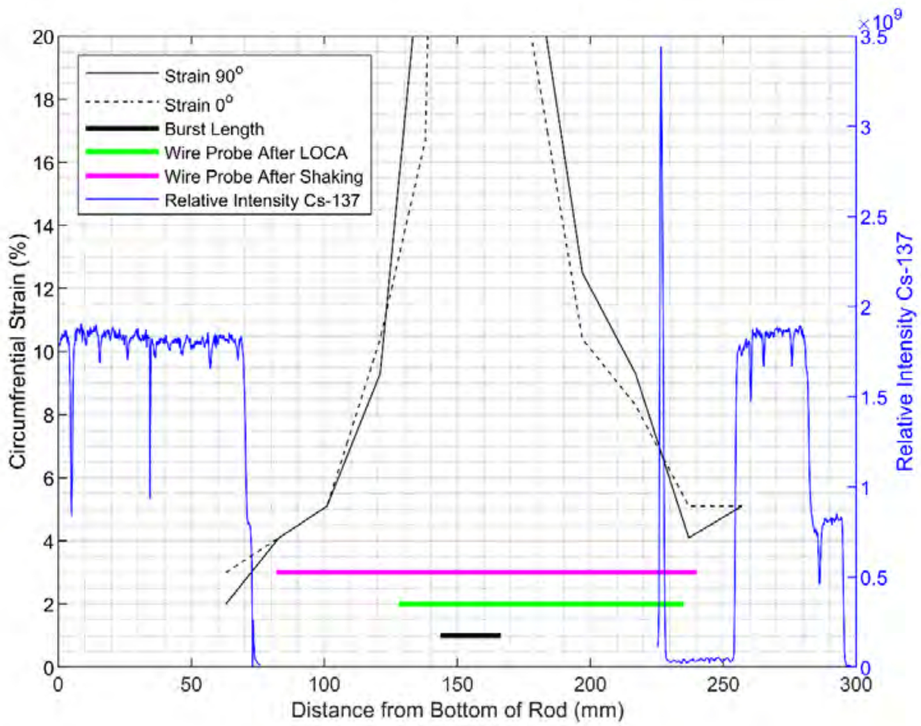
FFRD Thresholds



- NRC staff position is that fine fragmentation begins around a burnup of 55 GWd/MTU
- This is a simplification of complex processes in the fuel
  - Burnup is only a surrogate for microstructural changes
  - Other parameters (e.g., temperature) influence fragmentation behavior

Figure 3 from RIL 2021-13

FFRD Thresholds



- Relocation is limited below a cladding hoop strain threshold of 3%
- Thus, fuel rods with pellet-average burnups above 55 GWd/MTU that balloon and burst during LOCA are susceptible to fuel dispersal

NRC test #	Strain threshold, top (%)	Strain threshold, bottom (%)
189	6.0	3.0
191	6.0	4.0
192	5.0	4.0
193	1.0	4.0
196	3.0	5.0
198	4.5	9.0

Figure 4 and Table 1 from RIL 2021-13

Analytical Limits for Fuel Dispersal

- DG-1434 provides acceptance criteria for the dispersed fuel mass
  - No fuel dispersal for breaks < TBS
    - Can be addressed by showing no ballooning and burst for rods peak pellet burnup > 55 GWd/MTU
  - For breaks > TBS, either show there is no fuel dispersal or show that other criteria are met (see upcoming slides)
- Dispersed fuel mass should be calculated using an approved evaluation model and fuel dispersal models
  - Evaluation model should include the impacts of transient fission gas release

Plausibility of the No Dispersal Criterion

- NRC staff analysis performed around 2013 provided fuel dispersal estimates for 3 plant designs (Westinghouse 4-loop PWR, CE PWR, GE BWR/4) (see ML23086B272)
  - Based on current licensed burnup limits and fuel management practices
  - Using nominal (rather than intentionally conservative) initial conditions
  - No dispersal predicted for CE PWR or GE BWR/4
    - Core wide PCTs of 700°C and 500°C, respectively
- During the IE rulemaking regulatory basis public comment period, NRC staff received comments stating that it may be possible to show no dispersal using more realistic LOCA methods allowed for break sizes > TBS

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Methods for Estimating Dispersed Fuel Mass

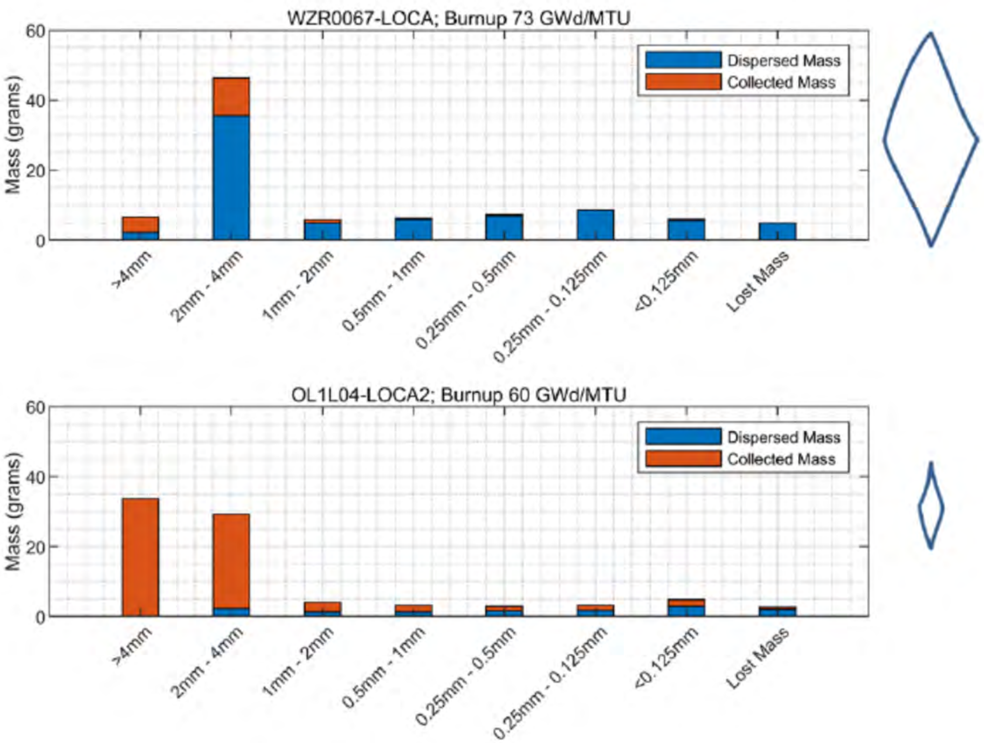


Figure A-4 from RIL 2021-13: Fragment size distribution for two SCIP-III tests, with relative burst opening size

- Fuel dispersal is impacted by many parameters, most of which are highly uncertain
  - Fuel dispersal PIRT identified "burst opening size relative to the fuel fragment size distribution" as high importance / low knowledge level
  - Other parameters like rod internal pressure also impact dispersal
- DG-1434 proposes a surrogate model for fuel dispersal
  - Avoids mechanistically modeling transport of particles through burst opening

Methods for Estimating Dispersed Fuel Mass

Figure A-6 from RIL 2021-13:
Mass fraction of fragments < 1 mm, with RIL Model A for comparison

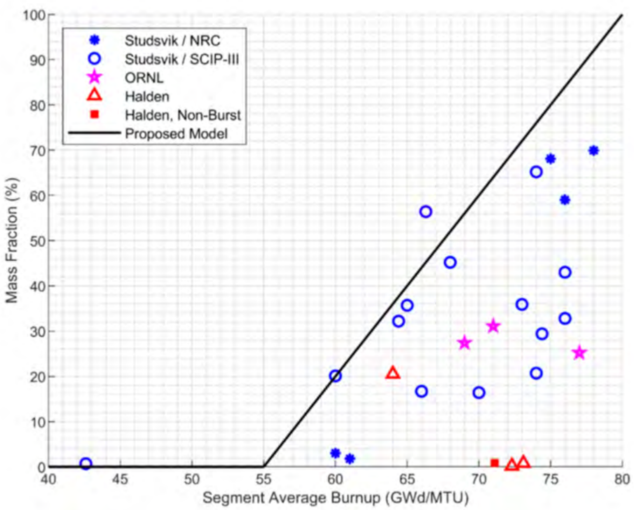
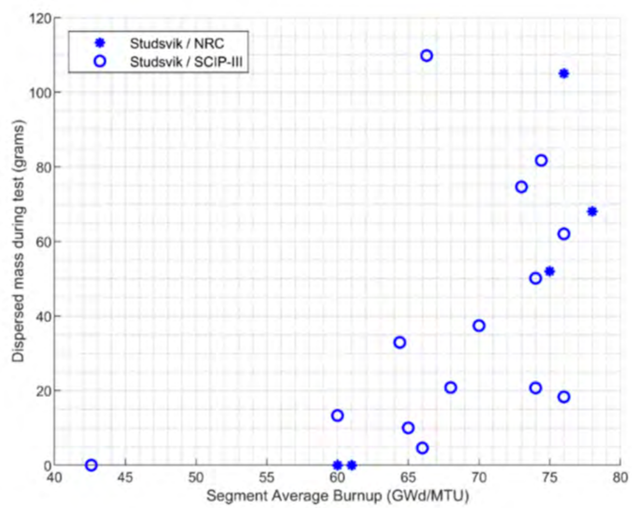


Figure A-1 from RIL 2021-13:
Mass of fuel dispersed during the test



- RIL 2021-13 provided surrogate models for fuel dispersal
  - Model A assumes that the mass of fragments below 1 mm is a good surrogate for dispersed mass
  - However, this does not mean that all dispersed fragments are less than 1 mm, nor does it mean that all 1 mm fragments disperse
  - Still, Model A is consistent with observations that dispersal increases with burnup

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Methods for Estimating Dispersed Fuel Mass

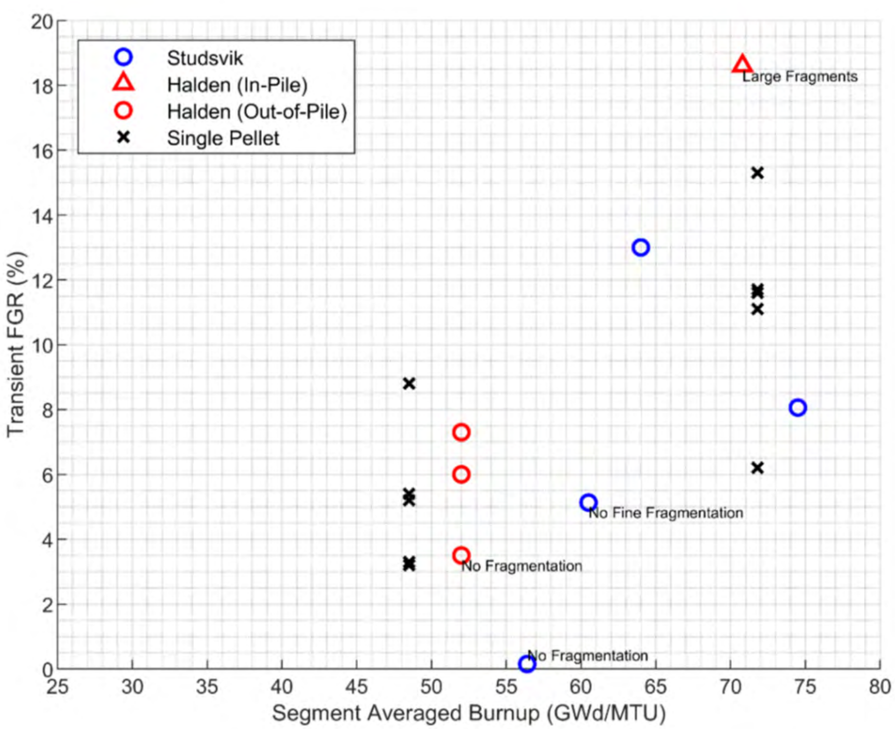
- DG-1434 recommends using RIL Model A to calculate fuel dispersal

$$\text{mass fraction} = \begin{cases} 0, & \text{BU} < 55 \\ 0.04\left(\text{BU} - 55\right), & 55 < \text{BU} < 80 \\ 1, & \text{BU} > 80 \end{cases}$$

  – Mass fraction should be multiplied by the mass of fuel in the region with >3% hoop strain
    - Can credit grid spacers to limit axial length of fuel susceptible to dispersal
  – Calculation should be performed for rods predicted to balloon and burst

	Difference between dispersal predicted by the model and dispersal observed in the experiment	
SCIP-III Test	A (mass, g)	A (%)
OL1L04-LOCA-2	29	314%
N05-LOCA	(10)	70%
VUR1-LOCA-1	(26)	76%
WZR0067-LOCA	(18)	75%
VUL2-LOCA1	34	169%
VUL2-LOCA3	142	874%
VUL2-LOCA4	99	259%

Methods for Estimating Dispersed Fuel Mass



- Methods for estimating mass of fuel dispersed should consider impact of transient fission gas release
  - Models should only consider gas release up to the point of ballooning and burst
    - Expected release at time of burst likely less than results shown in RIL 2021-13 due to burst temperatures being lower than test temperatures and impact of rod internal pressure in suppressing gas release
  - DG-1434 does not endorse any models but identifies potential starting points

Figure 7 from RIL 2021-13: Transient fission gas release as a function of burnup; note that peak temperatures for the Studsvik and single-pellet tests were between 1000 and 1200°C

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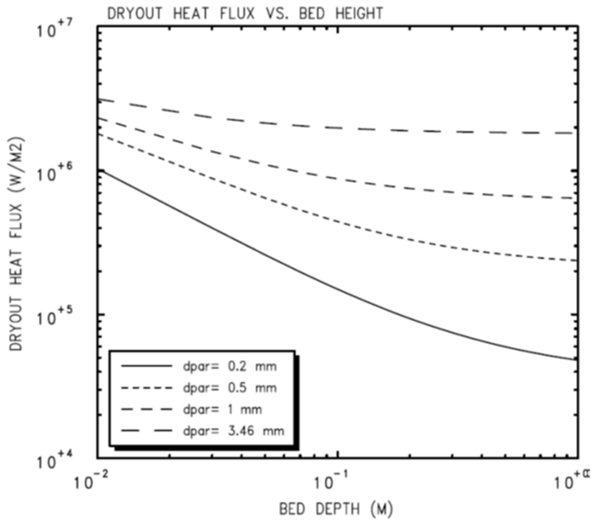
Fuel Particle Transport

- The fuel dispersal PIRT panel noted that transport of irregularly shaped particles in multi-phase flow is poorly understood
  - Panelists suggested addressing impacts of dispersal through simplified, bounding calculations, using engineering judgement about where particles may deposit
- DG-1434 identifies several potential dispersed fuel configurations to use when addressing fuel coolability
  - For example: on spacer grid immediately below cladding burst location, lower plenum, RCS piping

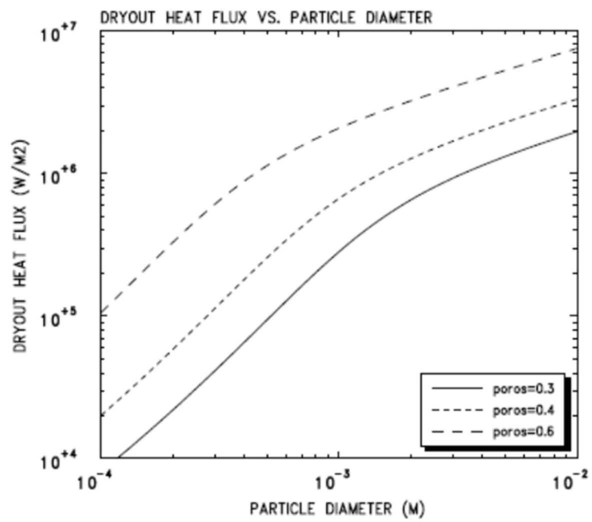
Fuel Coolability

- Analyses should address impact of dispersal on PCT and maximum local oxidation
  - Assuming fuel collects on spacer grid immediately below the burst location
- Analyses should verify that the dryout heat flux is not exceeded for particle beds on spacer grids and for other locations
  - Should use the 0-D Lipinski model to calculate dryout heat flux (i.e., maximum heat that can be removed from surface of particle bed)
- Analyses should perform calculations using range of conditions
  - Fuel particle size: 0.125 – 4 mm
  - Bed porosity: 20% – 40%

Fuel Coolability: Lipinski Model



porosity=0.4, P= 1.2 bar, T= 50 C



P=1.2 bar, T=50 C, bed height=53.2 cm

- OECD/NEA Working Group on Fuel Safety report on FFRD included dryout heat flux calculations
  - Showed dryout heat flux could be exceeded under some conditions (especially for small particle sizes)

Figs. 8.1-1 and 8.1-2 from the OECD/NEA Working Group on Fuel Safety 2016 report on FFRD

Fuel Long-term Cooling

- Analyses should demonstrate that adequate coolant flow is provided to remove decay heat from within the core and from fuel dispersed out of the core
- Analyses should verify that the dryout heat flux is not exceeded for particle beds
  - Should consider potential impact of coagulants or other debris that could reduce bed porosity below 20%

Dispersed Fuel Recriticality

- The fuel dispersal PIRT panel believed that recriticality of dispersed fuel is not a concern
  - Panelists also stated this could be demonstrated using existing tools and engineering judgement
- Staff performed simple analysis to address the potential for recriticality (ML24319A262)
  - Focused on simplified model for the lower plenum of Westinghouse 4-loop plant
  - Assumed all fuel was at 55 GWd/MTU (fine fragmentation threshold in DG-1434) and had initial U-235 enrichment of 8 weight percent
  - Did not credit soluble boron

Dispersed Fuel Recriticality

Pile Depth (cm)	K_{eff}	Fuel Mass (metric tons UO2)
20	0.725009 ± 0.000229	2.2
30	0.773866 ± 0.000261	4.0
40	0.800106 ± 0.000338	5.9
50	0.820710 ± 0.000286	7.9

CSAS-Shift model of lower plenum/fuel mixture (grey- steel / blue- non-borated water / green- Fuel/water mixture)

- Staff calculations show mass needed for recriticality far exceeds expected dispersed mass
  - For context: if all fuel in one grid space (~10% of the rod length) from all high burnup rods (~1/2 the rods in the core) dispersed, this would result in < 5 metric tons of UO_2 (for Westinghouse 4-loop plant)

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Dispersed Fuel Recriticality

- Staff analysis shows that recriticality is very unlikely
  - This is consistent with analysis performed by the OECD/NEA Working Group on Fuel Safety
  - Staff only performed quantitative analysis for one configuration, but based on engineering arguments recriticality is unlikely for other configurations
- DG-1434 states that licensees should demonstrate that the potential recriticality is addressed for their plant configuration
  - Licensees can use qualitative engineering arguments if dispersed mass is significantly less than the amounts in the staff calculation
  - At the same time, staff is working on providing stronger basis to resolve recriticality concern for the draft final rule package

Conclusions

- DG-1434 provides guidance for addressing the impact of fuel dispersal on ECCS performance
  - Guidance relies on use of more realistic LOCA methods and less conservative models from RIL 2021-13 to limit dispersed fuel mass
  - Guidance also includes methods to address impact of dispersed fuel on coolability
- Guidance is only one method for meeting regulatory requirement to maintain fuel coolability (50.46a(f)(1)(ii))
  - Industry can propose alternative approaches for NRC review

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Questions

Increased Enrichment Rulemaking for GDC-19 – Control Room and 10 CFR 50.67, Accident Source Term

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Division of Risk Assessment
Office and Nuclear Reactor Regulation
December 18, 2024

Topics

- Rulemaking Driver and Goal
- Background and History
- Regulatory Issues
- Rulemaking Proposal
- Rulemaking Approach

Rulemaking Driver and Goal

Legislative Driver:

- January 14, 2019, the President signed into law the Nuclear Energy Innovation and Modernization Act (NEIMA). Section 107, "Commission Report on Accident Tolerant Fuel," of NEIMA defines ATF as a new technology that makes an existing commercial nuclear reactor more resistant to a nuclear incident (as defined in section 11 of the Atomic Energy Act of 1954 (42 U.S.C. 2014)) and lowers the cost of electricity over the licensed lifetime of an existing commercial nuclear reactor.

Purposes:

- Facilitate the use of light-water reactor (LWR) fuel containing uranium enriched to greater than 5.0 weight percent uranium-235 (U-235).
- Developed in response to nuclear power industry interest to use fuel enriched to greater than 5.0 weight percent U-235 and deploy accident tolerant fuels (ATFs).

Staff Response:

- Evaluated areas within the regulatory framework and considered whether the current weight percent limits can be adjusted while maintaining reasonable assurance of adequate protection of public health and safety. Additionally, considered whether this rulemaking would support a more efficient review of licensing actions.

Background and History

Both GDC-19 and 10 CFR 50.67(b)(2)(iii) provide a specific dose-based criterion of 5 rem TEDE for demonstrating the acceptability of the control room design.

Represent a distinct layer of defense-in-depth that assumes a major accident that results in substantial meltdown of the reactor core with subsequent release of appreciable quantities of fission products.

Classic performance-based regulations which require that a licensee or applicant provide a control room habitability design using traditional deterministic radiological consequence analyses methods to judge the acceptability of the design.

Consequence analyses are also used to verify other regulatory requirements, guide maintenance activities, and serve as a guideline for performing 10 CFR 50.59 analyses.

Regulatory Issues

Assess Applicability in Current Environment

- Control room design criteria is limiting between the three acceptance criteria (EAB, LPZ, CR) for current enrichments and burnups.
- Development during the 1960s did not foresee how licensees are currently operating their facilities and managing fuel.
- The history of fuel utilization fleet has seen a gradual progression toward higher fuel discharge burnups and increased enrichments.
- There has been enough margin in the facilities' design- and licensing bases to accommodate the criteria, even for power uprates of up to 120 percent of the originally licensed steady-state thermal power level.
- Impact on Commission's comprehensive radiation protection and emergency planning frameworks.

Considerations of Control Room *Design* Criteria Impact is multifaceted

- Designer margin and operational flexibilities.
- Maturity of the regulated industry and compliance infrastructure.
- Maintenance activities and controlling actual operational exposure.
- 10 CFR 50.59 and low safety-significant licensing actions.

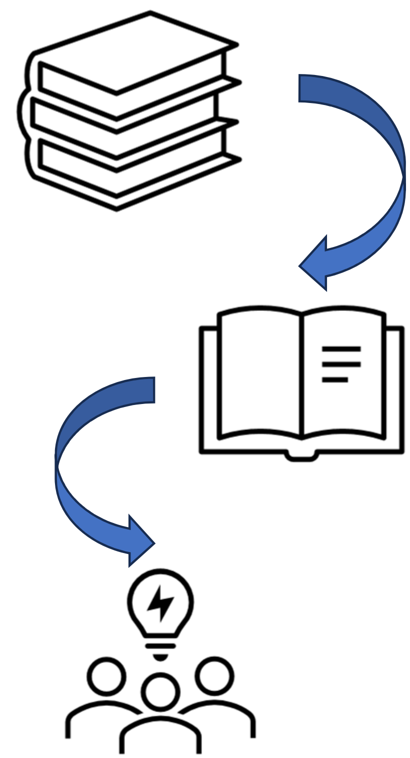
Radiological Risk Communications

- Design Criteria vs. Occupational Dose Limit.
- Radiation Protection and Emergency Response Frameworks.
- Health Physics First Principles and Radiation Epidemiology.
- Insights from Category 9 events.

Regulatory Issues (Cont.)

- The preamble for the 10 CFR 50.67 (64 FR 71990; December 23, 1999), final rule included the Commission's rationale for establishing 5 rem (0.05 Sv) TEDE as the GDC-19 numeric design criterion for licensees using an alternative source term. That rationale comprised the following:

> *"The criteria in GDC 19 were based on a primary occupational exposure limit.*
>
> *The use of 5 rem (0.05 Sv) TEDE as the control room criterion did not imply that this value would be an acceptable exposure during emergency conditions, or that other radiation protection standards of 10 CFR part 20, including individual organ dose limits, might not apply. This criterion was provided only to assess the acceptability of design provisions for protecting control room operators under postulated DBA conditions. The DBA conditions assumed in these analyses, although credible, generally did not represent actual accident sequences but were specified as conservative surrogates to create bounding conditions for assessing the acceptability of engineered safety features."*



Regulatory Basis Alternatives

Regulatory Basis document assessed three Alternatives. (88 FR 61986)

- *Alternative 1: No Action*
- *Alternative 2: Pursue Rulemaking to Amend the Control Room Design Criteria and Update the Current Regulatory Guidance Accordingly with Revised Assumptions and Models and Continue to Maintain Appropriate and Prudent Safety Margins*
- *Alternative 3: Update the Current Regulatory Guidance with Revised Assumptions and Models and Continue to Maintain Appropriate and Prudent Safety Margins*

Public Comments

Regulatory basis document sought comments on the alternatives proposed and asked two questions. (88 FR 61986)

Question 1:

Would the numerical selection of the control room design criteria be better aligned with regulations designed to limit occupational exposures during emergency conditions (e.g., 10 CFR 20.1206, "Planned special exposures," and 10 CFR 50.54(x)), or regulations designed to limit annual occupational radiation exposures during normal operations (e.g., 10 CFR 20.1201, "Occupational dose limits for adults," specifically the requirements in 10 CFR 20.1201 (a)(1)(i))? Please provide a basis for your response.

Question 2:

Would a graded, risk-informed method, to demonstrate compliance with a range of acceptable control room design criterion values instead of a single selected value, such as the current 5 rem (50 mSv) TEDE, provide the necessary flexibilities for current and future nuclear technologies up to but less than 20.0 weight percent U-235 enrichment? Please provide a basis for your response.

Assessment of Regulatory Basis Alternative 2 to perform rulemaking

- **<u>Option 2A</u>**—Amend the codified numerical acceptance value from 5 rem total effective dose equivalent (TEDE) to a new single value of 10 rem TEDE, with conforming changes to guidance.
- **<u>Option 2B</u>—Amend the codified numerical acceptance value from 5 rem TEDE to a range of values from 10 to 25 rem TEDE, with a graded, risk-informed, performance-based framework in guidance.**
- **<u>Option 2C</u>**—Amend the codified numerical acceptance value from 5 rem TEDE to a new single-value 25 rem TEDE, with conforming changes to guidance.

Proposed Rulemaking Language

- High-level Changes
  - Increase from 5 rem to 10 rem TEDE.
  - If additional operational flexibilities are needed beyond 10 rem TEDE, facility-specific risk profile or information can be leveraged to justify a higher numerical value up to 25 rem TEDE with is provided in DG-1425 (RG 1.183 Rev. 2).
  - Clarify the purpose of the control room design criteria and distinguish it from the radiation protection and emergency preparedness frameworks.
  - Consistence with other regulations containing either dose-based design criteria or radiation exposure limits.
  - DG-1425 adopts a method that develops a framework for a graded, risk-informed, and performance-based control room design criterion. Approach is consistent with SECY 98-144.

Proposed Rulemaking Language

Example of 10 CFR 50.67(b)(2)(iii) proposed language:

"(iii) **The necessary design, fabrication, construction, testing, and performance criteria for structures, systems, and components important to safety are** provided to permit occupancy of the control room under accident conditions without calculated radiation exposures in excess of **0.10 Sv (10 rem)** total effective dose equivalent (TEDE) **or a higher design criterion limit established in accordance with paragraph (b)(3) of this section** for the duration of the accident.

(3) The licensee may establish a design criterion limit higher than 0.10 Sv (10 rem) total effective dose equivalent (TEDE) but not greater than 0.25 Sv (25 rem) TEDE for compliance with paragraph (b)(2)(iii) of this section provided the licensee demonstrates that the specified limit is consistent with the plant risk-profile or commensurate with the risk of the plant.

Approach for Rulemaking for the Control Room Design Criteria

Policy and Regulation

- Flexibility within Commission Policy
- Radiation Protection and Emergency Response Framework

Evidence-based justifications

- Scientific Recommendations
- Radiation protection and radiation epidemiology

Ability to Provide Reasonable Regulatory Relief

- Reduce regulatory burden while maintaining safety and compliance
- Risk-informed and performance-based Rulemaking

Policy and Regulation – Flexibility within Commission Policy

- 10 CFR Part 20 puts into practice recommendations from the ICRP and NCRP. (56 FR 23360; May 21, 1991)
    - ICRP Publication 26, Recommendations of the ICRP (ICRP, 1977) subsequent ICRP publications.
    - NCRP Report No. 91, Recommendations on Limits for Exposure to Ionizing Radiation. (NCRP, 1987)

- From ICRP 26
    - Occupational exposure limit set to limit stochastic effects and prevent deterministic effects.
        - 5 rem/yr dose-equivalent to limit stochastic effects to an acceptable level.
        - 50 rem/yr dose-equivalent to all tissues except the lens to prevent deterministic effects.

- Both ICRP and Part 20 provide flexibility for planed special exposures.
    - ICRP proposal would have permitted a 15-rem dose in 1 year.
    - Part 20 condition theoretically possible to get a 10-rem dose in 1 year.
        - Concluded that an infrequent exposure of workers up to twice the occupational dose limit was adequately protective of radiation workers.

Finding: The system of dose limitation, as adopted by the Commission in 10 CFR Part 20, provides flexibility when considering risk-informing the dose-based control room design criteria.

Policy and Regulation – Radiation Protection and Emergency Response Framework

10 CFR Part 20, *Standards for Protection Against Radiation.*

10 CFR Part 50.47, *Emergency plans*.

Appendix E to Part 50—*Emergency Planning and Preparedness for Production and Utilization Facilities.*

10 CFR Part 55, *Operators' Licenses.*

10 CFR Part 50.54, *Conditions of licenses.*

10 CFR Part 50.155, *Mitigation of beyond-design-basis events.*

Finding: a range of regulatory dose-based occupational expose limits and design/siting criteria up to 25 rem TEDE. Understanding of their basis, purpose, and application helped inform IE rulemaking proposal.

Policy and Regulation –
Radiation Protection and Emergency Response Framework (Cont.)

Original GDC-19 (1971)	• At the time that GDC-19 was established in 1971, 10 CFR Part 20 limited occupational radiation exposure to 3 rem (0.03 Sv) whole body dose per calendar quarter, provided the total lifetime dose was verified not to exceed 5 rem (0.05 Sv) times the individual's age in years minus 18. Thus, a worker could receive a radiation exposure of up to 12 rem (0.12 Sv) in a given year.
10 CFR Part 20.1201, "Occupational dose limits for adults,"	• An adult worker could receive occupational radiation exposure of up to 10 rem (0.10 Sv) TEDE over a 12-month period straddling two calendar years.
10 CFR Part 20.1206, "Planned special exposures,"	• Permits an adult worker to receive doses in addition to, and accounted for separately from, the doses received under the limits specified in 10 CFR 20.1201 of five times the annual dose limits during the individual's lifetime, not to accumulate faster than 5 rem (0.05 Sv) TEDE in any one year. As such, an adult worker could receive radiation exposure of up to 10 rem (0.10 Sv) TEDE within a single calendar year period.

Policy and Regulation – Radiation Protection and Emergency Response Framework (Cont.)

10 CFR 50.47(b)(11)

- Establish the means for controlling radiological exposures in an emergency and states that the means for controlling radiological exposures must include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protection Action Guides (PAG).
  - EPA PAG Manual, recommends that doses received under emergency conditions should be maintained ALARA and, to the extent practicable, limited to 5 rem (0.05 Sv).
  - The guideline for actions to protect valuable property is 10 rem (0.10 Sv) where a lower dose is not practicable, the guideline for actions to save a life or to protect large populations is 25 rem (0.25 Sv) where a lower dose is not practicable, and exposures greater than 25 rem (0.25 Sv) may be appropriate for lifesaving or protecting large populations if the workers are volunteers who are fully aware of the risks involved.

Policy and Regulation –
Radiation Protection and Emergency Response Framework (Cont.)

10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance";

10 CFR 50.34, "Contents of applications; technical information";

10 CFR 50.67; and

10 CFR part 52.

- The upper range of the proposed numerical values would be consistent with the Commission's use of the 25 rem (0.25 Sv) TEDE limit primarily in regulations for power reactor siting to protect the public during emergencies.

- As discussed in the preamble for the final rule updating the NRC's siting criteria (61 FR 65159; December 11, 1996),

  "the Commission's use of 25 rem (0.25 Sv) TEDE does not imply that it considers it to be an acceptable limit for an emergency dose to the public under accident conditions, but only that it represents a reference value to be used for evaluating plant features and site characteristics intended to mitigate the radiological consequences of accidents in order to provide assurance of low risk to the public under postulated accidents."

Evidence-based justification – Scientific Recommendations

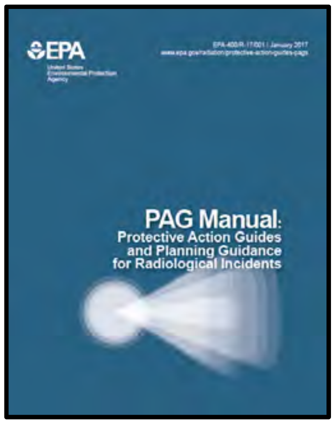
- Reviewed several source materials to understand the current recommendations from national and international organizations responsible for making recommendations for radiation protection standards.

- Purpose of this review was to determine whether reexamining the scientific and technical basis for the numerical value of the control room design criteria would be warranted.

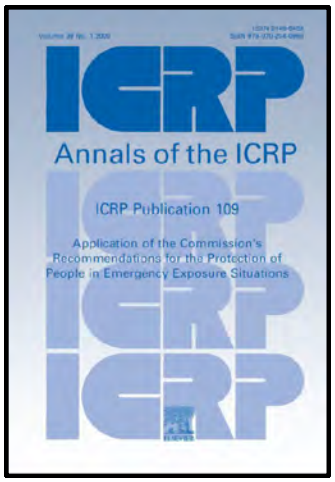
Finding - a range of international and national organization-based recommendations for radiation exposures for radiation workers under normal and emergency conditions up to 25 rem TEDE.

Source: Brock, Et al., NRC, White Paper, "Control Room Design Criteria and Radiological Health Effects (ADAMS ML23027A059)

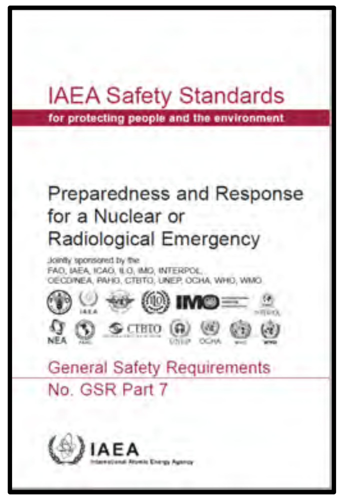
Evidence-based justification – Scientific Recommendations



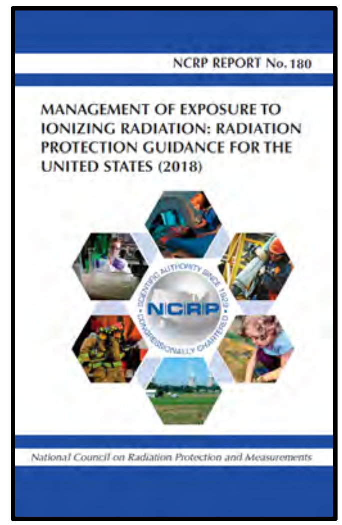
25 rem for lifesaving or protection of large populations



2–10 rem acute for emergency exposure situations



10-50 rem, 100 rad, acute depending on the severity of the actions needed



50 rad to 10 rem, depending on actions needed

Substantial and convincing scientific evidence of health effects following high-dose exposures. However, below levels of about 10 rem above background from all sources combined, the observed radiation effects in people are not statistically different from zero.

Evidence-based justification – Radiation protection and radiation epidemiology

- Proposed rule recommendations are firmly founded on modern radiation protection- and health effects knowledge.

- Deterministic Health Effects (rad)
  - Significantly below the threshold for observable deterministic health affects such as acute radiation syndrome and hematopoietic syndrome which occurs at doses around 70 to 100 rad respectively.
  - Far below the mean lethal dose of ionizing radiation without medical treatment, estimates to be approximately 300 to 500 rad.
  - Part 20 exposure limit set low enough to protect against deterministic effects.

- Stocastic Health Effects (rem)
  - Far below individual estimates of radiation risks for cancer mortality given the relatively short time frame exposures would be incurred.
  - Radiation protection and emergency response programs would actively monitor and manage occupational exposure before the 10 CFR Part 20 limit.
  - Traditional DBA radiological consequence analysis will continue to provide additional defense-in-depth from the 10 CFR Part 20 occupational exposure limits.
  - Continues to ensures a high level of protection is still provided, minizmizing long-term health impacts.

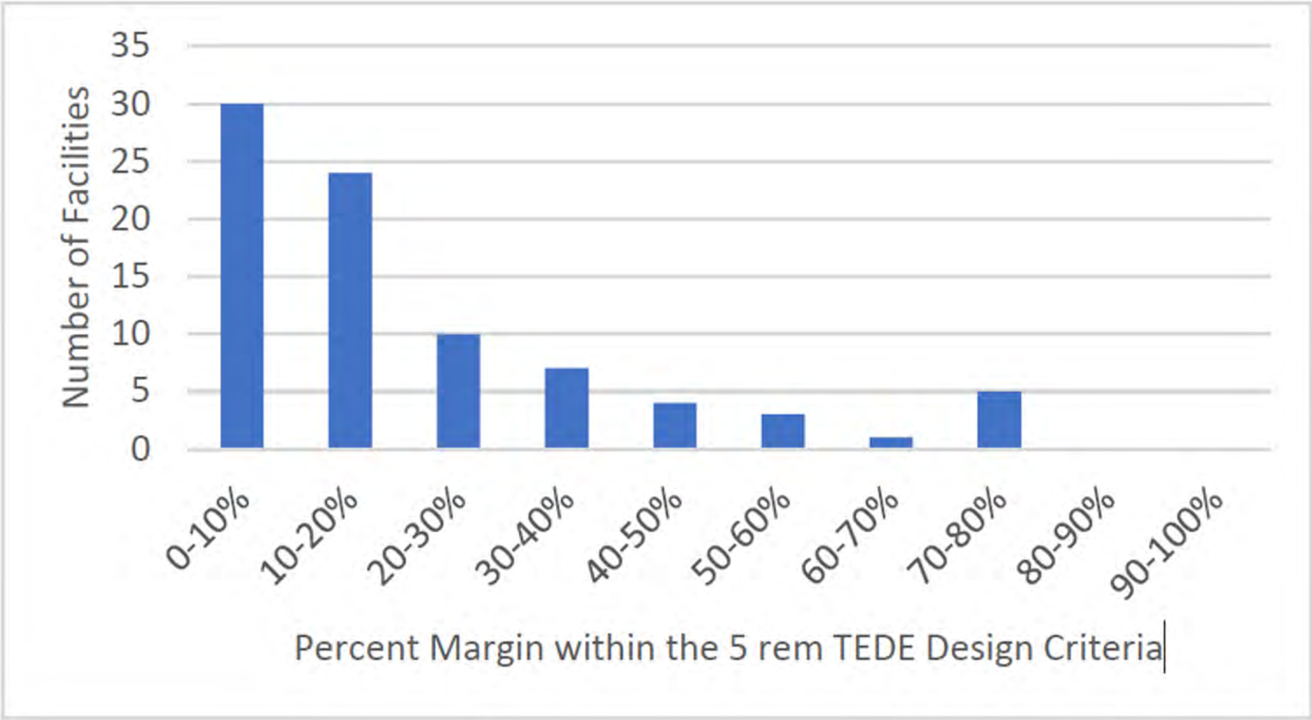
Ability to Provide Reasonable Regulatory Relief – Reduce regulatory burden while maintaining safety and compliance

- A very low design criteria value can result in an excessive amount of maintenance, leading to avoidable occupational exposure and unnecessary operational disturbances. Conversely, a very high value may allow for unacceptable degradation, potentially compromising overall safety and performance over time.

- Considerations of Control Room Design Criteria Impact is multifaceted:
  - Maturity of the regulated industry and compliance infrastructure.
  - Designer margin and operational flexibilities.
  - Maintenance activities and controlling actual operational exposure.
  - 10 CFR 50.59 and low safety-significant licensing actions.

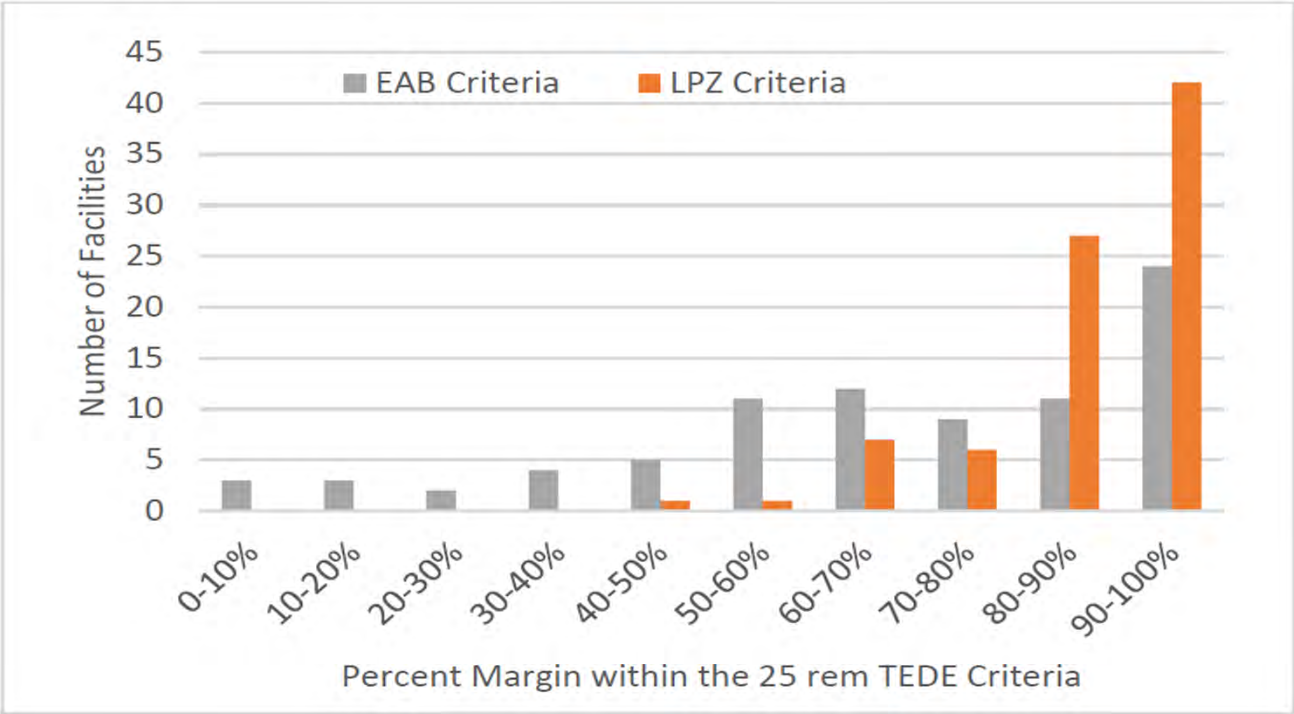
Finding: Adequate protection of public health and safety and occupational radiological safety can still be achieved at a higher, but still safe, control room design criteria performance level while balancing both dose-savings to workers and providing some regulatory relief to maintain operational flexibilities.

Ability to Provide Reasonable Regulatory Relief – Reduce regulatory burden while maintaining safety and compliance



Histogram of Licensees' Analysis of Record DBA Result Percent-Margin to Control Room 5 rem TEDE

Ability to Provide Reasonable Regulatory Relief – Reduce regulatory burden while maintaining safety and compliance



Histogram of Licensees' Analysis of Record DBA Result Percent-Margin to Control Room 5 rem TEDE

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

- Consider Commission-directed PRA-related policies which advocate certain changes to the development and implementation of its regulations using risk-informed, and ultimately performance based, approaches.
  - 1985 Severe Reactor Accident policy statement (50 FR 32138; August 8, 1985)
  - PRA Policy Statement (60 FR 42622, August 16, 1995)
  - **SRM-SECY-98-144, “Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulations” (ADAMS Accession No. ML003753601)**
- SRM-SECY-98-144 defines the terms and Commission expectations for risk-informed and performance-based regulation. Item 8, “Risk-Informed, Performance-Based Approach,” reads as follows:

  “… Stated succinctly, a risk-informed, performance-based regulation is an approach in which risk insights, engineering analysis and judgment including the principle of defense-in-depth and the incorporation of safety margins, and performance history are used, to (1) focus attention on the most important activities, ***(2) establish objective criteria for evaluating performance, (3) develop measurable or calculable parameters for monitoring system and licensee performance, (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes,*** *and (5) focus on the results as the primary basis for regulatory decision-making.”*

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

Break SRM-SECY-98-144 into Rulemaking- and Guidance Elements…

Element 1 – **Rulemaking** to design a rule which *establishes objective criteria for evaluating performance with developed measurable or calculable parameters for monitoring system and licensee performance.*

Element 2 – **Guidance** to design a regulatory framework which *provides flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes.*

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

Element 1

- Increase control room design criteria by a factor of 2, from 5 rem to 10 rem TEDE and range up to 25 rem TEDE with consideration of the plant-specific risk profile c risk information.
  - Numerical values are risk-informed based:
    - Commission Policy and Regulations for infrequent- and emergency exposures.
    - Recommendations from national and international and organizations responsible for radiation protection standards and guidance.
    - modern radiation protection- and epidemiology.
  - Performance-based aspect of rule retained within guidance which historically requires traditional DBA radiological consequence analyses thereby retaining staff's experience and licensee's licensing basis.
  - Flexibility and scalability incentivizes safety improvements if additional margin is needed beyond 10 rem TEDE where a lower facility-specific risk-metric allows for a higher and still safe performance criteria.

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

Element 2

- Establish in guidance, a framework for a graded, risk-informed and performance-based control room design criterion.
  - Enables a performance-based evaluation using traditional deterministic radiological consequence analysis methods within defined risk informed boundaries.
  - Boundaries are defined by acceptable radiation exposure guidelines for radiation workers during accident and emergency conditions and acceptable contemporary nuclear facility risk profiles using modern probabilistic risk assessment methods.
  - Provide flexibility when determining how to meet an established acceptance criterion in a way that encourages and rewards safety of the facility.
  - In practice, the method produces a framework that leverages in part, its safe design and operations to justify a higher control room design criterion with a lower plant-specific risk metric.
- Considerations:
  - Simple to understand and use.
  - Leverages well-known and understood methods and analyses.
  - Similar to other graded regulatory methods such as RG 1.174, SRP-specific DBA dose-based acceptance criteria, Frequency-consequence curves.

Source: Dickson, E., NRC, internal memorandum to K. Hsueh, "Method for a Graded Risk-Informed Performance-Based Control Room Design Criteria Framework," Washington, DC, July 2024 (ML24212A254).c

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

DG-1425 (RG 1.183 Rev. 2) Graded, Risk-informed and Performance-based Framework

- Risk-Metric Range: Regulatory Guide 1.174 CDF Criteria within five-bins.
- Criteria Range: 10 to 25 rem TEDE.
- Like similar licensing actions, approved higher licensing basis criteria is a “snapshot” in time until the licensee needs approval for an amendment (e.g. 50.59 criteria).
- Addressing framework defense-in-depth, safety margin, and uncertainty through license submittals.

Table 8. Guidelines for Control Room Location Based on a Graded, Risk-Informed, and Performance-Based Framework

Overall CDF	Graded Control Room Design Criteria (rem-TEDE)
CDF ≤ 1.E-5	25
1E-5 < CDF ≤ 5E-5	20
5E-5 < CDF ≤ 1E-4	15
CDF > 1E-4; or licensee not adopting the graded framework to determine acceptance criteria	10

Reference: DG-1425, Table 8

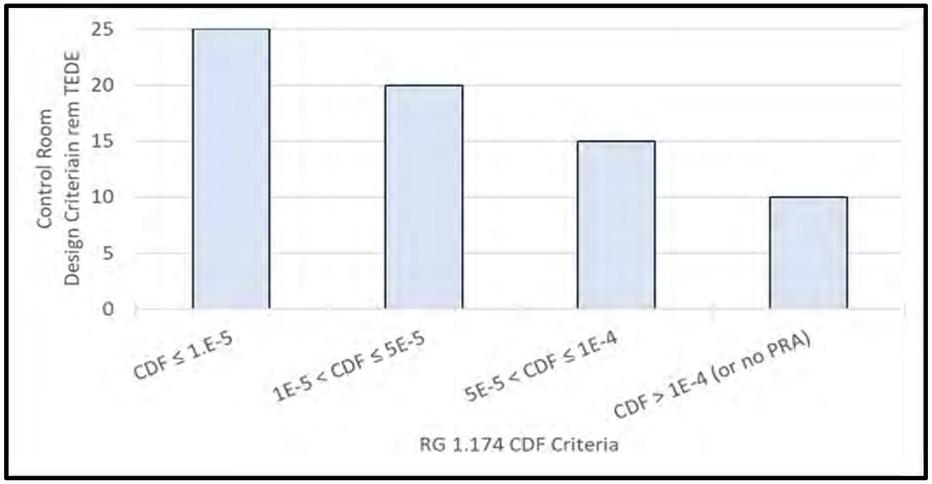


Figure representing DG-1425, Table 8 Graded, Risk-Informed and Performance-Based Framework

Using risk-information as a sliding scale to identify a higher control room design criterion.

Recap: Approach for Rulemaking for the Control Room Design Criteria

Policy and Regulation

- Flexibility within Commission Policy
- Radiation Protection and Emergency Response Framework

Evidence-based justifications

- Scientific Recommendations
- Radiation protection and radiation epidemiology

Ability to Provide Reasonable Regulatory Relief

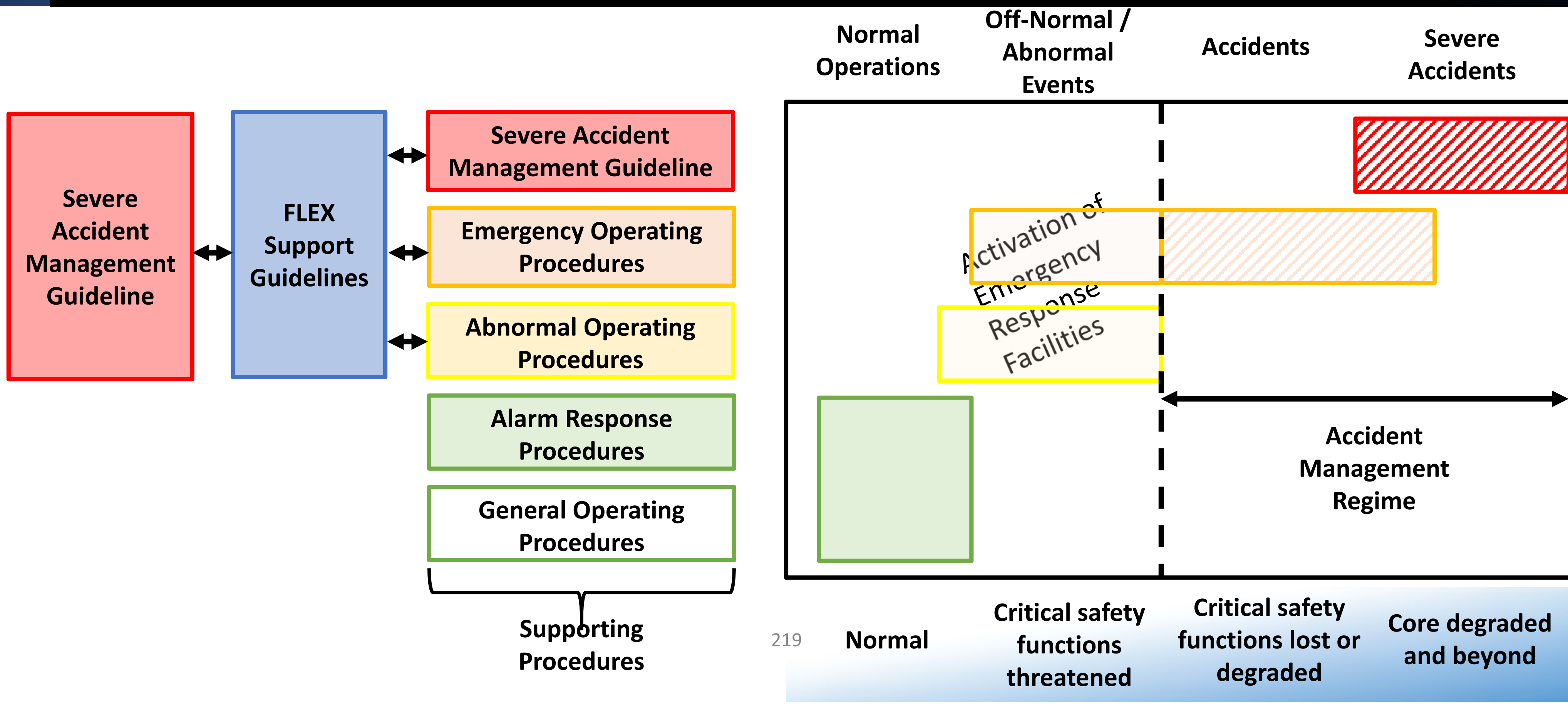
- Reduce regulatory burden while maintaining safety and compliance
- Risk-informed and performance-based Rulemaking

Closing

- The NRC recognizes the challenges that licensees face to retain margin within their licensing bases for the purposes of operational flexibility and the small amount of margin to the control room design criteria itself.
- The key driver behind the proposal to amend the control room design criteria is to facilitate increased regulatory efficiency and consistency while continuing to provide adequate protection of public health and safety.
- Proposed rule and supporting regulatory guidance executes Commission SRM-SECY-98-144 defines the terms and Commission expectations for risk-informed and performance-based regulation.
- This rulemaking effort would also support increased consistency within the Commission's comprehensive radiation protection and emergency planning frameworks.
- Consistency among these regulations, 10 CFR 50.67 and GDC 19 would provide operational flexibilities to further limit actual occupational exposures while also realigning the numerical value as a design criterion with a potential reactor accident of exceedingly low probability.

Backup Slides

Policy and Regulation – Radiation Protection and Emergency Response Framework (Cont.)



Class 9 Accidents

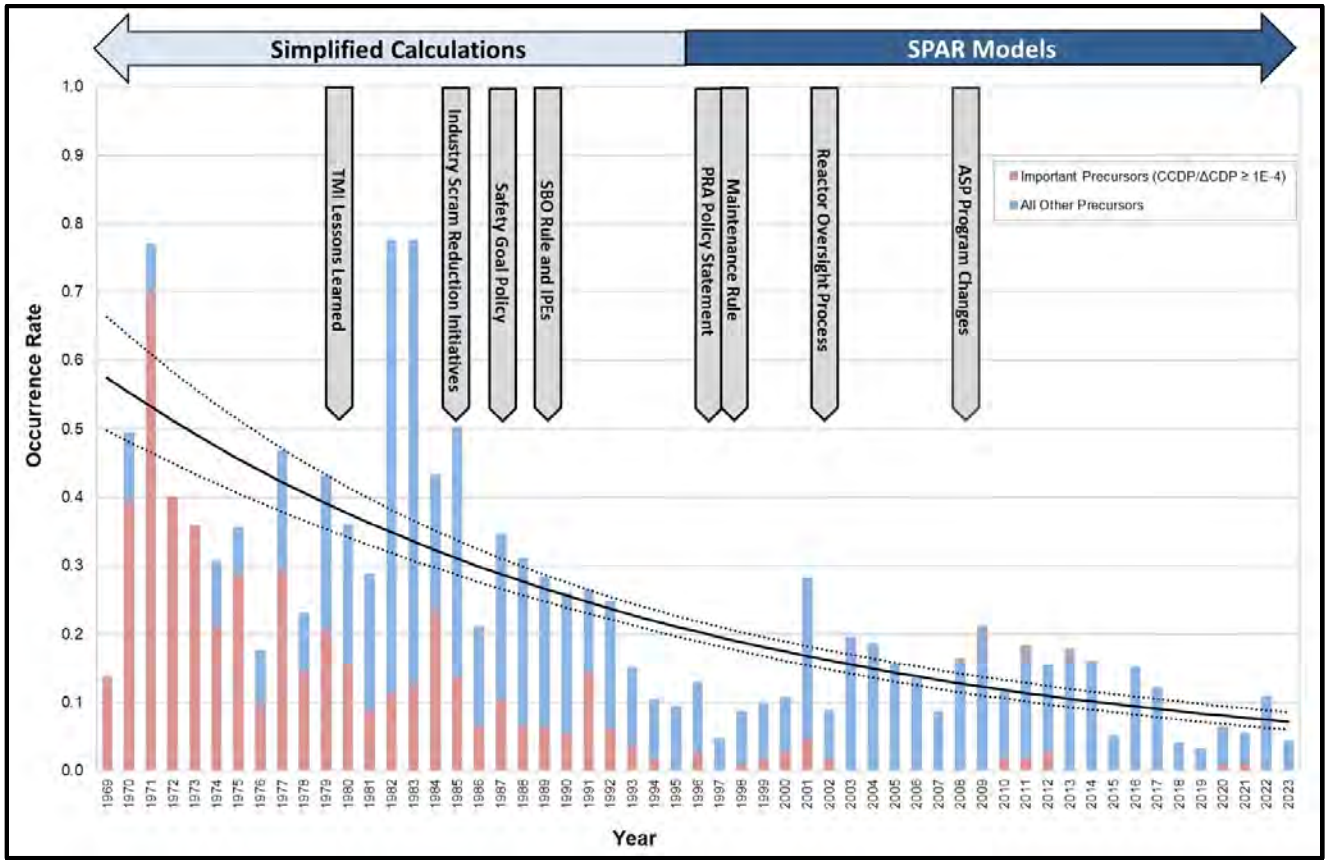
NRC Assessments of Severe Accident Risk

- 10 CFR Part 51, *Environmental Protection Regulations for Domestic Licensing and Regulatory Functions,* implement the National Environmental Policy Act (NEPA).
  - Consideration of the costs and benefits of severe accident mitigation/design alternatives and the bases for not incorporating severe accident mitigation design alternatives.
- 10 CFR Part 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*, requires an application to include a supplement to the environmental report that complies with the requirements of Subpart A of 10 CFR Part 51.

Severe Accident Events of Western Designs

- Three Mile Island
  - Maximum whole body dose received by an actual individual during the TMI accident in March 1979, which involved major core damage, was estimated to be about 0.1 rem. (61 FR 65159; December 11, 1996)
- Fukushima Daiichi
  - Maximum effective dose was 67.9 rem, six received effective doses greater than 25 rem, and 168 workers between 10 to 25 rem. (UNSCEAR, 2021)
  - Internal contamination was attributed to the severe working conditions and the inadequate implementation of protective measures (e.g. improper use of respiratory protection, iodine thyroid blocking measures, actions that resulted in inadvertent ingestion of radionuclides), due primarily to the lack, or ineffectiveness, of training. (IAEA, 2014)

Accident Sequence Precursor Program



Source: *U.S. Nuclear Regulatory Commission Accident Sequence Precursor Program Summary Description*, Appendix B: *Historical Precursor Occurrence Rates* (ML24177A020)