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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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685TH MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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WEDNESDAY

MAY 5, 2021

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The Advisory Committee met via Videoconference, at 2:00 p.m. EDT, Matthew W. Sunseri, Chairman, presiding.

COMMITTEE MEMBERS:

MATTHEW W. SUNSERI, Chairman

VICKI BIER, Member

DENNIS BLEY, Member

CHARLES H. BROWN, JR. Member

VESNA B. DIMITRIJEVIC, Member

GREG HALNON, Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Member

DAVID A. PETTI, Member

JOY L. REMPE, Vice Chairman

PETER RICCARDELLA, Member

ACRS CONSULTANT:
MICHAEL CORRADINI

DESIGNATED FEDERAL OFFICIAL:
DEREK WIDMAYER

ALSO PRESENT:
CYRIL DRAFFIN, USNIC
SCOTT MOORE, Executive Director, ACRS
QUYNH NGUYEN, ACRS
WILLIAM RECKLEY, NRR
JOHN SEGALA, NRR
MARTIN STUTZKE, NRR
NANETTE VALLIERE, NRR

P R O C E E D I N G S

2:00 P.M.

CHAIR SUNSERI: It is 2:00 p.m. Eastern Time. The meeting will how come to order. This is the first day of the 685th meeting of the Advisory Committee on Reactor Safeguards. I'm Matthew Sunseri, chair of the ACRS.

I will now call the roll and confirm a quorum and that clear communications exist. Normally, we would start with Ron Ballinger, but he's not available to attend this week and he has an excused absence, so I'll go to Vicki Bier. And I know that Vicki had contacted me and thought her availability might be a little uncertain this afternoon, so sounds like she's not available either. And that is an excused absence also.

Dennis Bley.

MEMBER BLEY: I'm here.

CHAIR SUNSERI: Charles Brown. Charles Brown. Vesna Dimitrijevic.

MEMBER DIMITRIJEVIC: Here.

CHAIR SUNSERI: Greg Halnon.

MEMBER HALNON: I'm here.

CHAIR SUNSERI: Walt Kirchner. Walt Kirchner. Jose March-Leuba. Jose? Dave Petti.

MEMBER PETTI: Here.

CHAIR SUNSERI: Joy Rempe.

MEMBER REMPE: Here.

CHAIR SUNSERI: Pete Riccardella.

MEMBER RICCARDELLA: Here.

CHAIR SUNSERI: And myself. So let me check here. One, two, three, four, five, six, seven. We barely have a quorum.

Walt, are you on yet?

MR. CORRADINI: I thought Walt said he was coming on at 3 p.m.

CHAIR SUNSERI: Oh, that's right. Yes, he did talk to me about that. That's another excused absence. So how about Jose? Jose?

MR. NGUYEN: I just let him in, so I think he's here.

CHAIR SUNSERI: Jose, are you talking about Jose?

MR. NGUYEN: Correct. Charlie is also showing up.

MEMBER BROWN: I have got to leave my desk here for a minute. I am here.

CHAIR SUNSERI: Okay, you're there.

MEMBER BROWN: I've got to take care of myself. I'll be back in about three minutes.

MEMBER RICCARDELLA: Walt shows as being here, too.

MEMBER BROWN: I just logged in just now.

CHAIR SUNSERI: All right. Well, we have a starter quorum now. We'll go ahead and get started. Maybe by the time we get through with the introductions everybody will be present.

MEMBER BROWN: Okay, I'll answer present when you call me out.

CHAIR SUNSERI: That's fine. All right, so let me just divert a little bit here before I continue. I want to take a moment and recognize the fact that former Commissioner Lyons passed away last week and you've likely seen reports on the multitude of technical accomplishments he made. I unfortunately never had the privilege to work with him during his times as a Commissioner and made several drop-ins and visits with him and I found him to be a very brilliant technically, and of very sound character.

And I know there's other members on the committee that knew him much better than me and I'll just pause at this moment to see if anyone wants to say anything.

MEMBER REMPE: Sure, Matt. This is Joy. And yes, I was fortunate enough to work with Pete when he was at DOE. Again, there are many kind and wonderful things one can say about Pete, but what always amazes me was his low-key manner and civility and had a very kind of way interacting with people which I greatly appreciated.

CHAIR SUNSERI: Yes, I totally agree. He was a special person. He's going to be missed by our industry.

I think Walt had a close relationship with him as well and wish he was here to say something, but anyway. All right, anybody else? Thank you for that.

The ACRS was established by the Atomic Energy Act as governed by the Federal Advisory Committee Act. The ACRS section of the U.S. NRC public website provides information about the history of the ACRS and provides documents such as our charter bylaws, Federal Register notices for meetings, letter reports, and transcripts of all full and subcommittee meetings including all slides presented at the meeting.

The committee provides its advice on safety matters to the Commission through its publicly available letter report. The Federal Register notice announcing this meeting was published on April 7th, 2021, and provided an agenda and instructions for interested parties to provide written documents or request opportunities to address the committee. The Designated Federal Officer for this meeting is Mr. Derek Widmayer.

During this week, the committee will focus on the following for the remainder of most of the day or the remainder of the day, we're going to take up an interim letter report on 10 CFR Part 53, rulemaking for licensing of advanced reactors. There will be some staff presentations and then we will get into report preparation following that.

Tomorrow morning, we will begin with a White Paper on Fusion which is an informational briefing.

Regarding Agenda Item 5 for tomorrow, uprated NuScale standard design approval application update, NuScale has requested that this item be removed from the agenda. NuScale plans to provide an update at a date closer to when the standard design approval would be submitted. The purpose of the meeting was to discuss NuScale's 250 megawatt thermal standard design approval regulatory engagement plan and which proposed four phase review process.

Interested members of the public may access the regulatory engagement plan in ADAMS and I'm going to give a number here so I'll pause just a second so that you can ready yourself to copy. The ADAMS number is M as in Mike, L as in Lima, 21047 A as in Alpha, 475. That's the engagement plan and the four phase review process proposal is M as in Mike, L as in Lima, 21112 A as in Alpha 183.

I apologize for any inconveniences this may have caused by changing this agenda this way.

The other topics that we will take up in under the general topic of report preparation and other committee business and that is the NuScale control room staffing plan. This is a letter report that carried over from the last committee meeting. And this will be our number one priority to get this letter report out this meeting. And we also are updating our bylaws as an action item from our retreat. We will work this item in as time permits.

As far as the interim letter on Part 53, there's a lot of work that still needs to go into this letter and I talked to Dennis. My goal will be to have a read in of the draft letter today and then try to get agreement on recommendations and conclusions by the end of the week. If we can complete the letter by the end of the week, that would be the stretch goal, but right now I don't know if that's going to be achievable or not, but we can get at least through the read through and the agreement on the draft recommendations and conclusions that would be a good position to be in.

A phone line, a phone bridge line has been opened to all members of the public to listen in on the presentation and committee discussions. We have received no comments and only one request to make oral statements from a member of public regarding today's session. And this is a request from USNIC and that will come during the comment period following the Part 53 presentation.

There is also an opportunity for public comment and we have set aside time in the agenda for comments of members of the public attending this meeting. Written comments may be forwarded to Mr. Derek Widmayer, the Designated Federal Officer. A transcript of the open portion of the meeting is being kept and it is requested that speakers identify themselves and speak with sufficient clarify and volume so that they may be readily heard. Additionally, participants should mute themselves when not speaking.

Now one small change you noticed during the roll call is we have two new members that have been appointed to ACRS since our last full committee meeting. And I want to welcome Vicki Bier and Greg Halnon to the committee here. I'm going to do a little bit of introductions here. I'll put my camera on for this. Vicki is not here, so I'll save this for tomorrow.

But Greg Halnon is an independent nuclear industry consultant who has more than 40 years of experience in the nuclear industry. Mr. Halnon has expertise in all aspects of nuclear plant operations, as well as quality standards, security maintenance, and engineering processes. He currently holds a professional engineering license in two states and has held two senior reactor operator licenses during his career.

Mr. Halnon earned a Bachelor of Science degree in engineering from the University of Central Florida with emphasis on mechanical and thermal hydraulics. And it is also worthy to note that Greg is a life member of the American Nuclear Society.

So Greg, welcome to the committee. If you have anything you want to say before we get started?

MEMBER HALNON: Thank you, Matt. Real briefly. I just appreciate everybody's welcoming and it didn't take me long to very much appreciate the quality of people, both on the staff and ACRS. And I really look forward to interacting through this appointment. So I appreciate the time.

CHAIR SUNSERI: And I'm not doing something right here with my attendee list, but I see somebody has their hand up and I don't know who that is, so whoever has their hand up, you have the floor.

MEMBER MARCH-LEUBA: It might be me, sorry. This is Jose. I'm back. I've been taking every single thing so yes, I am back.

CHAIR SUNSERI: Could you hear us when we were doing the roll call?

MEMBER MARCH-LEUBA: No, I couldn't. It was a long story, but I'm back.

CHAIR SUNSERI: Okay, all right. No problem. All right, all right, well, that's good. We have a strong quorum now.

All right, so that is all for the introductions and opening remarks. I'll open the floor to the committee. Any member have anything you want to say before we get into the agenda?

All right, well, then at this point I will turn the floor over to Dennis Bley for the interim letter report on 10 CFR 53.

MEMBER BLEY: Thank you, Mr. Chairman.

This is not dealing directly to the letter; the staff at the back to give us a presentation. I have asked them to do another overview especially on Subparts B and C for members who were not able to listen in all of our subcommittee meetings.

I've asked the staff to do this as quickly as reasonable to do that review. There were also some areas that came up at our April 22nd meeting where some additional staff's expertise would have been in response to questions from the committee and I've asked them to go back over some of those issues so that they're planning to do that.

After we get through the staff presentation, you may know that USNIC has asked for a chance to speak and Mr. Cyril Draffin will then provide comments to us on their behalf. And then we'll do a read through of the letter that's been put together to support this thing and we'll try to make sure we get through all of that before the end of the day.

So at this time, I'm going to turn it over to staff.

John Segala, did you want to begin or somebody else?

MR. SEGALA: Yes, thank you. Yes, this is

John Segala, Chief of the Advanced Reactor Policy Branch in the Office of Nuclear Reactor Regulation. And consistent with the Nuclear Energy Innovation and Modernization Act, or NEMA, we are committed to developing a technology inclusive, risk informed, and performance based regulatory framework for a wide range of advanced reactor designs and publishing the final Part 53 rule by October of 2024 in accordance with the Commission's directed schedule.

We are committed to a regulatory framework for advanced reactors that achieves the goals of the Commission's advanced reactor policy statement and the NRC's principles of good regulation. We are having extensive stakeholder engagement where we release preliminary rule language to solicit feedback to better inform the staff's proposals and to ensure a shared understanding of what will be included in the final rule.

As we are considering changes to the previously released preliminary rule language, we want to ensure that we have appropriately considered the feedback we have received from all stakeholders including the public, industry, standards development organizations, trade groups, non-governmental organizations, and the Advisory Committee on Reactor

Safeguards.

Since we are at the early stages of the rulemaking process, the draft preliminary rule language will remain open for discussion as the staff works towards providing the Commission a proposed rule. We are here today in the fifth of many ACRS meetings we will be having this year to seek ACRS feedback on the NRC's development of Part 53 preliminary proposed rule language for advanced reactors.

We previously briefed the ACRS Subcommittee in January on the first set of preliminary rule language in Subparts B and F, in February on Subparts C and D in March, where stakeholders shared their insights and we discussed the structure and logic of Part 53, key guidance needed for Part 53 and Subpart E on construction and manufacturing.

In April, our last meeting where we discussed the second iteration of the preliminary rule language in Subparts B and C and the key elements of the Part 53 framework in order to set the stage for the ACRS full committee meeting today.

Today, we plan to provide the full committee an overview of the Part 53 structure and the preliminary rule language for Subpart B and C. We also plan to provide additional information to help answer questions brought up during the April subcommittee meeting. We understand that ACRS plans to develop an interim letter report following this full committee meeting and we are looking forward to hearing any insights and feedback from the full committee today, as well as the conclusions and recommendations in the ACRS interim letter.

This completes my opening remarks. And I can turn it over to Bill Reckley or Bob Beall.

MR. RECKLEY: Yes, this is Bill Reckley. Did you have something, Dennis?

MEMBER BLEY: No, I'm conferring, but I would like to hear from you though.

MR. RECKLEY: All right. We can go to the next slide.

So on Slide 2, as John mentioned, we're going to go over the overall structure. We're not going to spend too much time on that. There seem to be an general understanding and at least for now a general support of the overall structure. Then we're going to look at Subpart B on the safety requirements and Subpart C design and analysis.

As Dennis mentioned, there's a fair amount of material to try to go through in a couple of hours. So we're going to go relatively quickly. If there's a need to stop and pause and go over some things, that would be understandable. But some of it also, some of the specific topics like the probabilistic risk assessment, some elaboration on our plans to use PRA within Part 53, we have added a few slides and Marty Stutzke will be doing that presentation when we get into Subpart C on design and analysis.

But for now, if we go to slide -- the next slide. John mentioned this in the background already. We had, as part of overall plans for advanced reactors, considered a rulemaking even back in 2016, as we were laying out our strategies. We were -- then events kind of overtook us with the passage of NEMA and signing that into law in 2019 and that law specifically told us to develop a framework through rulemaking to address advanced reactors and so that changed our schedule a bit and is largely the reason we're here today.

If we go to Slide 4 --

MEMBER REMPE: Bill?

MR. RECKLEY: Yes.

MEMBER REMPE: This is Joy. I thought of something when we last met that Dennis and I discussed later. And I think Dennis said he didn't hear what I heard and I'm not, maybe I got confused. But I asked you a couple of questions last meeting about mostly gatekeeper, you were trying to do with Part 53 giving folks a little more flexibility because they are an advanced reactor with increased reliance on passive and inherent safety features.

So what if someone comes with something that's a Superphenix, you know, Clinch River thing that doesn't have any passive, well, not many, passive or inherent safety features. What's the gatekeeper -- because I thought you'd said well, if we see something like that and it's not meeting the requirements, we're going to impose some additional requirements on them. It sounds kind of fuzzy to me.

When do you decide it's got sufficiently increased reliance on passive and inherent features? That sounds a little fuzzy.

MR. RECKLEY: And to be clear, NEMA gave a number of criteria and some of it was related to passive or inherent safety features. Other of the criteria within NEMA had to do with cost of electricity, fuel utilization, or waste yields, reliability, proliferation, increased thermal efficiency which most or at least many of the nonlights would engender, or the ability to use for non-electric applications like hydrogen production. So within the NRC's advanced reactor policy statement, the focus was, as you mentioned, passive inherent safety benefits.

Within NEMA, there's a number of considerations that could qualify one to be quote an advanced reactor. So we had in our rulemaking plan acknowledged that light water at a minimum, light water SMR, Small Modular Reactors, and any non-light water reactors, so a generation for technology be it a Superphenix or a medium size fast reactor like PRISM or some of the micro reactor designs being considered now. Any of those would have been falling into that category most likely.

The question had become what about large light water reactors, the Generation III+ kind of technologies. Our original thinking was they may or may not come into play. So at this point, you know, there will be many more that would go through the gate than would be stopped by the gate. Maybe I can just put it that way, if that answers the question?

MEMBER REMPE: What's the gate? I recall a long time ago that the Commission said well, the current fleet is safe enough and advanced reactors don't have to be safer because the current fleet is safe enough. And we didn't give them increased flexibility, so it would sound to me is what you're saying is everything gets in or is there some place where if they meet the safety criteria everybody is in, right?

MR. RECKLEY: Yes, pretty much. The only question that really NEMA raised when you look at those criterias and this is just from my point of view, the only question raised is it said advanced reactors other than those under construction at the time of the act. And the only construction under way at the time of the act was AP1000.

So that's why we were questioning whether Gen III+ might be included, but all of this will be kind of brought out as we finish out this rulemaking and agree on the scope. But there won't be very many technologies excluded in my mind based on the criteria that Congress included which went well beyond passive or inherent safety features and included things like fuel utilization, non-electric uses and so forth.

MEMBER REMPE: Then I guess my next question is why do you keep saying these things are going to be safer because you just had told me everybody is in?

MR. RECKLEY: In the advanced reactor policy statement, first of all, I hope we don't overuse that they are safer, as you mentioned, and as we'll get into as we go through Subparts B and C. The way I like to put it is they provide their safety through different mechanisms and light water reactors include a certain amount of reliance, for example, on mitigation, including emergency planning, siting restrictions, and so forth.

One of the goals of Generation IV reactors and we've heard from stakeholders that this remains true is to lessen the reliance on things like siting and emergency planning as a safety measure and as it was discussed in the advanced reactor policy statement, to ensure safety more through the design of the facility and the use as you've mentioned a couple of times, the use of passive and inherent safety features.

So when you look at it as an integrated assessment of the safety of any particular plant down the road, the overall safety in terms of protecting public health and safety will be at least as good as what we have now for the light water reactors, but more importantly how you get that safety might be different.

MEMBER REMPE: It might be different, but it doesn't have to be different.

MR. RECKLEY: It doesn't have to be different.

MEMBER REMPE: Yes. I think we need to keep this in mind because making a lot of assumptions about oh, they're going to do more things with passive and inherent, but unless there's a gate to keep that and what is more reliance, there isn't any.

MR. RECKLEY: Right.

MEMBER REMPE: So that's like of a fiction here, it's all everybody's dream, but anybody can get through the gate is I guess what I'm learning a little more explicitly.

MR. RECKLEY: And another way to put that is, for example, we're allowing, we're trying the way we're writing the rule, to allow for the use of less reliance on emergency planning if it can be justified. But we're not requiring that there be no reliance on emergency planning, right?

So going to exactly what you said, they could achieve safety through some of the same measures, siting restrictions, and emergency planning, and so forth as the current fleet. And we're trying not to preclude it, but we're also saying if in the design process you can justify that you don't need to rely on those measures, then we're trying to build in the flexibility to say that you've proved your point through the design process and you don't need those additional measures that were imposed for light water reactors.

MEMBER REMPE: Thank you. I appreciate this long discussion on this, but I think it's good to lay it out on the table.

MR. RECKLEY: And it's actually, thank you, because it's kind of important as we go to the next -- we can go to the next slide on Slide 4.

MEMBER BROWN: Bill, Bill?

MR. RECKLEY: Yes.

MEMBER BROWN: I just want to echo Joy -- go backwards again.

MR. RECKLEY: Okay.

MEMBER BROWN: You make the comment, this is what bothered me the whole time on these advanced reactors, they're safer, they're passive. We've heard that continually and they're uranium. It's got to be fissioned in order to produce power. And you won't produce power for large populations, a lot of power in many cases, most cases. So you've got all the same -- the same pot is cooking, it's just being handled differently.

How in the world can you ever get away from emergency planning zones and site boundaries and all that other kind of stuff and/or dosage requirements? I don't see how you could ever -- it's all the same stuff. All we're doing is adding more toxic means in most circumstances, lead bismuth, sodium. I'd love to have one of those plants go melt down somewhere.

So the idea that they're passive and that makes them safer just means it's less complex to be -- to make sure the plant shuts down. So I hate the advertising of we're going to get rid of emergency planning zones and everything else. It's just an observation. I had to see it used. So I just needed to say my piece as well.

I'm not criticizing you, Bill.

MR. RECKLEY: No, that's fine. And again, we're just trying, and we'll get into this a little later in the discussion, trying to take this integrated look and say when would you rely on certain provisions and when might you justify that you don't need to rely on those provisions. We'll get into that a little later as we go along.

MEMBER PETTI: Just for the record, your

discussion of how safety is achieved is exactly what the Gen IV International Forum and the experts internationally thought about as they rolled out the leading Gen IV concepts. So I see a lot of consistency with that, thank you. Thanks.

MR. RECKLEY: Okay, thanks, Dave. Okay, so if we can go to Slide 4. This is our overall structure for Part 53 and how we arranged it into a number of subparts.

And most of the discussion with the subcommittee thus far has been on Subparts B and C. But the general arrangement is that Subpart B was intended to layout the safety goals, the safety objectives, the criteria that would be used and the need to identify safety functions.

And then the other subparts were basically organized along the lines of a project lifecycle and were intended to, for example, under design and analysis, Subpart C say, what is the contribution that design and analysis provides to make sure that any particular plant, any particular design meets the safety objectives and meets the safety criteria.

And then likewise, what's the role of citing, construction, operations and how would that be carried into retirement. So that was the nature of the main technical subparts, B through G.

Then in addition, there were subparts related to licensing that we're currently developing. Those are Subparts H and I. And then administrative reporting, other general provisions, that would be in Subparts A and J.

But again, most of the focus with the Subcommittee, and with external stakeholders, has been on Subparts B and C, up to this point.

So, if we got to Slide 5, another way to lay this out for the Subcommittee. We had gone through the individual chapters. For the sake of time I didn't do that today, but this just lays out the same thing from the graphic in kind of more of a chart or a table of contents with the Subparts A through J.

Subparts B and C are highlighted because we want to spend more time talking about those today. In red is just some of the, some notes on particular subparts.

And in particular, on Subpart B, under safety criteria, some of the discussion topics as included. Our organization of the requirements into the first and second tiers, or categories, of safety objectives.

The use of as low as reasonably achievable within the criteria related to normal operations. And how you achieve defense-in-depth.

And then under the design and analysis, some discussion on how we propose to have the design criteria addressed and the role of probabilistic risk assessments that we'll get into in a few minutes.

So, I had not planned on spending much more time. If we can just go back to Slide 4 for a second.

I hadn't planned to spend much more time on the overall structure or organization of Part 53, unless there are specific questions.

MEMBER BLEY: Yes. This is Dennis, Bill. Charlie brought up a point and argued it, and it's similar to, Charlie, it's similar to what you wrote down for, and delivered it in the last meeting last week.

And essentially it boils down to, you cannot assume that new reactors coming in will be safer. And the objective to that assumption.

And it seems to me, although some of the hype makes that assumption the approaching you're taking doesn't allow for it. You can't assume it, you have to show that you meet the level of safety. You can comment on that or not?

MR. RECKLEY: Well, I think as we're trying to, as we get into Subpart B, on the criteria, the other thing that we will emphasize is that based on past Commission decisions, the highest level criteria remain the same. We haven't proposed, for example, and we'll get into the discussion on the health objectives, but we haven't proposed to use different health objectives, we're using the same ones from the advance reactor policy statement.

Again, how you achieve those objectives might differ from design to design. In terms of the plant design, there is going to be reliance on different barriers and technologies based on the type of reactor.

And again, we're laying out the possibility that if any designer or licensee wanted to use mitigation measures, the comparable up to what light water reactors do, then we're not precluding that. So, I'm not sure I addressed your question, but --

MEMBER BLEY: That's fine, Bill.

MR. RECKLEY: Okay. All right, so if there is no other discussion of the overall structure, we can get into Subpart B. Yes, thank you.

And then just go onto Slide 7. One point that I'm not sure we emphasized in the Subcommittee meeting that I did want to just revisit and emphasize by having a slide is that we have said in various papers, including the rulemaking plan sent up to the Commission, SECY-32, that we were planning to build Part 53 based on the activities that were ongoing at that time or that we had completed shortly before then. Such as Secy-19-0117. Which I won't read that long title, but the shorthand of that is licensing modernization project.

And NEI 18-04, the ACRS looked at that paper and at the associated reg guide, Reg Guide 1.233. But I just wanted to reiterate that our plan was to take such a risk-informed approach. And that's what was communicated to the Commission in those papers and what was accepted within the SRM for both the rulemaking plan and SECY-19-0117.

So, if we go to Slide 8. This goes largely to both Charlie and Joy were mentioning. The nature of a reactor is that it's making fission process. It's how it makes its energy, and as a side product it's making fission products. And that's the hazard.

And this graph basically shows that as the black inventory shape. And the nature of reactor safety is basically to provide barriers to the dispersion of those radionuclides to the environment.

And in some cases, as we talked about a couple of times, if you cannot preclude the dispersion then you might have mitigation measures on the outside. Such as restricting where you can cite them and/or providing protective actions, such as the sheltering or evacuation in nearby populations. And so, all of those things considered are what determines the risk to public health and safety.

We first used this graphic, or a graphic that was similar to it, in SECY-19-0117 to try to describe how within the risk-informed approach we were reflecting in that paper, considers things like mechanistic source term and a more integrated approach.

And so, you will see the other paper cited there is the functional containment paper, SECY-18-096. Where if, to simplify a little bit, for light water reactors the general approach has been generally to pick bounding, challenging kind of events for each barrier.

And so there would be challenges to the cladding and then there would be challenges to the reactor coolant system and challenges to the containment. They may be different events. And the most challenging event basically establishes the design requirements on each barrier.

Under this you're still looking, you largely are going to have the same or types of barriers, but you're taking a more, you're looking at more event scenarios and taking an integrated approach to looking at each scenario. And that is reflected, again, the functional containment paper, in the licensing modernization paper, as another way to basically look at ensuring that appropriate barriers are in place to the release of radionuclides.

And we'll get into this as we go into the discussion a little more on licensing bases events and Marty's discussion on the use of the probabilistic risk assessments.

But the, so, let's go on to Slide 9. This is a slide we used during the Subcommittee meeting.

And we look at Part 53 in our construct, and that overall structure, one of the things to keep in mind, just to keep the terminology straight as we go through Subparts B and C, is this --

THE OPERATOR: If you're on the bridge line please mute your phone. Please mute your phone.

MR. RECKLEY: Okay, thank you. The kind

of hierarchy is shown in the chevrons that we start in Subpart B laying out the safety criteria, then we require safety functions as a means to satisfy those criteria.

And then when we get into Subpart C we'll talk about design features. Which is the hardware, the structure systems and components needed to carry out the safety function. And then functional design criteria, which are the more specific things associated with the design feature to make sure that it will support meeting the safety functions and the safety criteria.

So over in the white boxes, the functions are things like what barrier is needed, what cooling might be needed to maintain a barrier.

The design feature would be specific structures and systems and components, pumps, heat exchangers, control rods, whatever the function and then whatever the design feature you're using.

And then the design criteria would be things like leak rate, reactivity insertion rates, cooling capacities, more specific engineering parameters associated with components. Like pumps and heat exchangers and so forth.

So this is the layout of Subpart B, with the first, or major focus being the safety criteria and the safety functions. And then in addition we have specific requirements on assessing the unplanned or licensing basis events, ensuring defense-in-depth and the protection of workers.

So, another way to characterize Subpart B is these are the, Subpart B is the what. What are we trying to accomplish. And that's, again, meeting the safety criteria, supplying the safety functions.

Subpart C, and all the other subparts, get into the how. What are the design features. In the parentheses we start to address things that we will put in requirements in operations related to human actions. What are the role of personnel and so forth. So, the other subparts talk about the how.

So, if we go down one more. I repeated this graphic, again, in Slide 10, and enhanced it just a little bit to bring in an example of how, in the past, we've used such an exercise of going from functions to features to functional design criteria in a specific reactor type. And this is the MHTGR. Modular high temperature gas reactor.

As we'll talk about in a minute, this exercise was also gone through in order to arrive at the general design criteria or the advance reactor design criteria that are included in either Appendix A to Part 50 or in Reg Guide 1.232.

But you can see that the safety function, as we've currently defined it in Subpart B, and we'll get to the language in a minute, starts off with the ultimate goal of limiting the release or radionuclides. And then identifies what other functions are needed to carry that out.

And that is, those functions that are needed to protect whatever barriers a designer is choosing to accomplish those functions. By and large, that's going to be the fuel, some reactor system. Whether it be the fuel encased in the cladding or some kind of pressure boundary. And then in some cases, an additional structure as a last barrier.

But for MHTGR it was identified, and we gave the example during the Subcommittee meeting, that you had heat generation or reactivity, heat removal. This is decay heat removal systems, emergency systems. And then for that time frame, when MHTGR was being considered, they identified chemical interactions as a different function.

But within our system, if those are the required safety functions, then Part 53 would then also say, within Subpart C, that we'll get to, you have to identify the design features that are going to accomplish those functions and then you identify additional engineering parameters to show that they can support that.

And if you go back and look at the MHTGR, and then to some degree, how that was carried through next generation nuclear plant, you can see how this has fed into the approach that we're proposing for Part 53.

So, as the ACRS mentioned in your letter on SECY-19-0117, that methodology and the methodology that you're seeing in Part 53, is really an evolution over the last 30 years. And so, some of what we're going to do today is a little bit of history to kind of fill in where we're getting this.

So, if you go to Slide 11, this is basically the same slide again. The top of this slide is right out of MHTGR and NGNP documents that show how you go down and determine those, what are your required safety functions.

And then I just added on to that figure for Part 53 space. In Subpart C you would do the design features and the functional design criteria in order to fill out the detail on how you were doing something like removing decay heat.

So, if we go to Slide 11. I mean 12. Thank you. One of the things that we talked about, and think is true, is that if you look at the general design criteria and the exercise that was done in the late '60s to develop the general design criteria for light water reactors, and then even more recently three or four, well, from three years ago with the issuance of Reg Guide 1.232 on the advance reactor design criteria, the same exercise that we just described in going from safety functions to design features to functional design criteria, was largely what was done for the, to develop the specific requirements for light water reactors that's reflected in the GDC.

And so, if you look at the safety functions in the left in the first column, you'll see reactivity control, fluid systems for heat removal and containment systems. Those align pretty closely to the fundamental safety functions of reactivity heat removal and containment.

Or if you're familiar with another one, the three C's, control, cool and contain. So those principles were laid out in the GDC.

And then for light water technologies, they basically filled in some specifics that became the individual GDC. So, we think it's a exercise that is similar.

In order to support Part 53, which is intended to be a technology inclusive approach, what we're building into Part 53 is a requirement to go through this methodology. And every designer or applicant ultimately would have to do this in order to come up with how they're going to perform the functions and what design features they're going to rely on to do that.

So, it is, to some degree, what we're proposing to do on Part 53, and when we get to the actual language in Subpart B, is to replace a fairly perspective list of technical requirements with a methodology to accomplish the same thing.

And when we were talking to the Committee during the review of the reg guide and the SECY paper on the licensing modernization project, we had the same discussion of looking at these as a methodology and a requirement to go through this exercise, versus having a prescriptive list because the regulator, or someone else, had already done it.

So, a useful exercise, if you have time, is to really look, for example, at the reg guide on the advance reactor design criteria, Reg Guide 1.232, and look specifically at the MHTGR. One of the technologies that's addressed in the advance reactor design criteria is the MHTGR.

I point out that it's particularly interesting because the MHTGR was really, from a design and licensing process, the genesis of much of what we're talking about, in terms of licensing modernization, and even moving forward into Part 53. And so, those folks that were involved in that, at the time of the NGNP, were looking at the ARDC and translating and doing this exercise.

And so you'll see, through the MHTGR ARDC, some degree of how this plays out. And it somewhat proves the point, at least to me, that the methodology can get you to basically the same place.

If we now can go to 13. It just finishes out the rest of the GDC, in terms of the other safety functions. Fluid systems or cooling and containment.

And then, we can now start to get into, okay, we have one more slide and then we'll get into the Subpart B actual language.

So the next slide, Slide 14, again, just tries to do some comparison of what people are more familiar with, which is the light water or Part 50 construct. And some of the changes or alternatives that are being looked at in Part 53.

So, we talked earlier, the safety criteria are basically the same. We're using the same reference values. The 25 rem at the exclusionary boundary. We're using the same QHOs.

Albeit, the QHOs don't show up specifically in Part 50, but as we've talked about, they are used in Part 52. Specifically under Chapter 19 of the SRP. I think Marty will talk about, more about that when he goes through some of the PRA discussions.

Within the design and analysis area, the design basis events are similar. But under Part 50, given the way that Part 50 was developed, it's more prescriptive. It's more conservative.

It includes, for example, and we're going to have specific slides on the single failure criterion, so I have it highlighted, but I didn't want to spend much time on this slide.

Under Part 53 it's more, it still includes a deterministic DBA in terms of a test of safety related equipment. It still performs that function of having a deterministic DBA.

It's probably a little more, it's a little less conservative under Part 53. And the reason for that is the next bullet. Is that, under Part 50, beyond design basis events were kind of ad hoc under Part 50.

And under Part 53 you have a whole category, a new category of events, in which you're doing a methodical assessment. And coming up with design and programs and operator actions needed to address the events down in that category. We will also talk about that when Marty does the PRA discussion.

The special treatment for non-safety related, but safety significant SSCs.

MEMBER BLEY: Bill?

MR. RECKLEY: Yes, Dennis.

MEMBER BLEY: I want to stop you just a second. I agree with what you said up there but if one looks at your 53.450, Paragraph F, analysis of design basis accidents, it uses the words conservative and the other typical words. But it doesn't really define what they mean.

See, what they mean right now is defined from the SRP in Chapter 15, and you don't have any definition here. Which means some people are kind of feeling empty that there is no requirement.

MR. RECKLEY: Yes. And that might be a

great point to clarify or enroll language or guidance provide.

The reason I say it's somewhat less conservative is that the event that's being analyzed under Part 50, and in particular, for example, treating the double ended guillotine break. The frequency of that particular event, if you looked at it from a PRA standpoint, might move it under Part 53 down to a beyond design basis event.

And it would still need to be addressed but it might now show up as the design basis accident as it was, for good reason, for the light water reactors when it was developed using the process developed for Part 50. The evolution of Part 50.

So, in terms of like the thermal hydraulics, and some of the conservatisms that are built in to making sure that if you're using a particular correlation or something like that, then that would be, that's, when we say a conservative analysis under, in our Part 53, Section 450, that's what we were referring to when we say conservative.

You know, you'd have to make sure that the actual modeling including the appropriate conservatisms in the DBA. Some of this we'll get into as we discussed the specifics, I think.

But again, the reason I highlighted single failure, that came up during the Subcommittee meeting, but we have a couple of slides specifically on that. And also looking at the combinations of failures within the PRA, and the two topics are related, as we'll see as we get into the discussion.

So, Slide 15. So now we're going to get into the specific language. On Slide 15, this is the language that we established, that the objectives are to limit the possibility of an immediate threat to public health and safety, and then appropriate measures considering risks.

There was a discussion during the Subcommittee meeting on whether we would need to define those terms more.

And I just wanted to point out, and have highlighted here, that from our perspective, though the meaning of those terms, if you just read them as they're written, might lead to questions. But the last sentence there is meant to clarify what we mean by that. And these safety objectives shall be carried out by meeting the safety criteria identified in the subpart.

So to translate that, what do we mean by an immediate threat to public health and safety, is an event that would lead to 25 rem over two hours at the exclusionary boundary or over the duration of the event at the low population zone boundary. So that's what we're equating to be an immediate threat to public health and safety.

And then as we get to the second tier, what do we mean by appropriate, considering potential risk to public health, that's the QHOs. So, we do think that sentence, hopefully, puts in context what we mean by the high level objectives.

If we go to 16, we go into start talking about the first tier. And the language, as I just said, was that the dose, largely from a, well, it's two parts.

Part A is normal operations. And we include, within the first tier safety criteria, the 100 millirem from Part 20. That's the annual dose from normal effluence.

And then more focus is on the unplanned events. And again, we use the same reference values. And as we'll talk about under, as we go down, this analysis is a DBA type analysis, only relying on safety related equipment. And it will show that the dose at the EAB, or low population zone boundary, is less than 25 rem over the duration of two hours or over the whole course of the event.

So if we go then to Slide 17, this is somewhat repetitious of what I just said. Twenty-five rems, same reference values we've historically, only relied on safety related equipment demonstrated by a deterministic type DBA.

It also is the vehicle for which we ensure an appropriate protection against external hazards. Again, that's largely consistent with how it's done now where the safety related equipment is protected against design basis seismic events or floods or other hazards.

And then one last point is, we're going to carry this through, as I mentioned, through the whole rest of the subparts. And it shows up again, for example, under what is the equipment that would be handled and controlled, tightly, through technical specifications. It would be the equipment needed to satisfy this first tier.

So the desire there would be to be always able to say that the plant is meeting that first level goal of not presenting an immediate threat to public health and safety.

And then as we get into the second tier, you'll see a parallel where we try to take a risk-informed approach. And also carry it through the design, the construction and operations where we basically have the same table for non-safety related but safety significant equipment, and the need to define special treatment for all of those things throughout the lifetime.

So, if we go onto Slide 18.

MEMBER HALNON: Excuse me, Bill?

MR. RECKLEY: Yes.

MEMBER HALNON: Bill, this is Greg Halnon and I just wanted to comment on the immediate aspect of this. And we don't have to have a lengthy discussion, but by putting the term immediate in the rule itself, it gives it a very temple emphasis as opposed to the way you're describing it, at least in my mind, is more emphasis on consequence opposed to the tempo aspect of it. So keep that in mind.

And when I read it I see a tempo, urgent tempo aspect to it. And the way you described it, at least in my mind is, more of a consequence or an ultimate consequence of an event that could be very long as opposed to intermediate thing. So, anyway, that's my opinion there.

MR. RECKLEY: Yes. And we'll look at the language to see if the word immediate, and where we got the word immediate threat to public health and safety is actually from the case law on technical specification and what is the appropriate content to tech specs.

And knowing that we wanted to carry that threat all the way through, and just looking at how things had been characterized, tech specs, and again, since it's been what we've regulated, large light water reactors, it was put in the terms of an immediate threat to public health and safety.

And so, maybe we can look at that. We'll look at that language as we go through the future iterations. I understand what you're saying though.

So, if we go then to Slide 18 it lays out the second tier of criteria. And it's been much discussion, but for normal operations, normal effluence, we have kept that they should be kept as low as reasonably achievable.

We're looking at future wording to tie it into Part 20. And also into an appropriate relationship with environmental protection agency requirements under Title 40.

Under unplanned events, again, it's been a lot of discussion, but the highlight texts is basically the existing quantitative health objectives that the immediate health effects or prompt fatalities would be less than five and ten million. And the risk to, of latent health effects would be less than two and one million years.

Again, based, that is the existing QHOs just put out into words.

So, going on to Slide 19, just sensitive, I received so much attention from stakeholders. We have slides that we gave to the Subcommittee that just, noting that many stakeholders did not believe ALARA meant, a range of comments from ALARA shouldn't apply to advance reactors down to, ALARA didn't need to meet, in Part 53, because it was already addressed in Part 20, to some proposing to keep it more or less as we had proposed it.

Which is the same as it is provided in Part 50. Specifically, although it's old, Appendix I to Part 50.

So, our iteration has been, as we discussed on the previous slide, to keep the ALARA requirements in place. And looking forward, we did it for occupational exposures, as well we for normal effluence.

So if we go to Slide 20, the other area that got a lot of discussions with stakeholders during the roll out of the preliminary language was on the use of the QHOs. And again, a range of comments from, don't include the QHOs, don't use the numerical aspects of the QHOs and try to put it into more general wording, to some who were in favor of basically using them as we had proposed in the preliminary language.

And our iteration has been to basically keep them as we proposed in the first iteration. You saw the language, we continued to refer to them as the primary metric for unplanned events in the unlikely and very unlikely event categories.

So, if we go then to Slide 21.

MEMBER DIMITRIJEVIC: Hi, this is Vesna Dimitrijevic. I just want to make comment on your previously slide. Where you said the QHO is a well-established measuring using risk-informed, I would challenge that because the QHO are not directly ever used in the risk-informed, just substantive measures.

MR. RECKLEY: Well --

MEMBER DIMITRIJEVIC: You know, CDF. And no one ever in the application looks back to QHOs. I mean, they are originally used to deny those CDF, but they're based on the couple very significant assumptions which have never been checked.

So I would say the subsidy objectives are well established, but not QHO. By any means.

MR. RECKLEY: Okay.

MEMBER DIMITRIJEVIC: They're based on the conservatives and things like that. Like a conservative change of like 30 percent to the aspect, yes. Nobody ever goes back to QHOs.

MR. RECKLEY: And we'll have some slides that Marty will talk about the QHOs and their assessment. I will say more recently, before advance reactors I worked in the area of the Fukushima response, we used the QHOs.

When we were making determinations on things like whether boiling water reactors should have filters on the release, we were using the QHOs. When we were looking at the assessment of whether we should expedite the fuel, spent fuel transfer from poles to casks, we used the QHOs.

So, yes, light water applicants have not traditionally used the QHOs because, in large part, surrogate measures have been developed. There have been, in recent cases, the use of QHOs and decision making.

But again, I don't want to get ahead of ourselves, Marty is going to talk about that a little bit. Or we have backup slides when Marty talks about probabilistic risk assessments.

So, I understand what you're saying. Not disagreeing that for light water reactors the use of surrogates, such as CDF and large release frequencies have been used instead of QHOs. Marty can better address the derivation of those surrogates. So, we'll get to that in a few, in a few minutes.

MEMBER DIMITRIJEVIC: All right.

MR. RECKLEY: So, Slide 21, goes to somewhat of a caution, if you will, that one of the reasons we need a metric, and it would have to be, as all the aspects of Part 53 has to be technology inclusive, but we have to have a fairly high level metric, but well-defined metric, within the safety criteria, is because we are proposing to use those metrics, again, throughout on how you procure equipment on which quality assurance requirements would be applicable.

Down into operations of how would one define what the reliability targets are for equipment, and that comes from the probabilistic risk assessment. And a metric for that analysis, which we're currently proposing to use the QHOs.

So, in the absence of a well-defined

metric, you make things like requiring an applicant to define reliability targets for equipment that much harder, if you don't have a metric to use for that purpose. So that's all I wanted to say on Slide 21.

If we go to Slide 22, we had some discussion during the Subcommittee meeting on the safety functions. Basically we lay it out that the primary function is the retention of radionuclides.

And then a requirement for additional safety functions to be identified, and again, the previous slides I had gone through on how to go through that exercise. And largely what was done in the late 1960s to develop the GDC was similar.

We gave examples of heat removal, heat generation and chemical interactions. The ACRS Subcommittee mentioned they thought reactivity should be mentioned, and we'll commit to including that.

There was, I think as some other members mentioned, we thought that was somewhat addressed but maybe less clearly by saying heat generation. We don't mind including reactivity as a specific example.

The notion of requiring or identifying those particular safety functions as requirements versus a requirement to assess and identify them, our concern is that, and it's highlighted down in the text box, one of the things that an applicant will need to do is identify safety functions for every major inventory.

And for some technologies, the other inventories, like waste gas, can be comparable in challenge to the reactor system itself. And so, under the way we have it worded, hopefully they would need to identify safety functions for that waste gas system.

And reactivity would not, or maybe even heat generation would be not as important. But they would have to identify, for that waste gas system, what are the safety functions needed to retain the radionuclides.

As the secondary concern, and we'll talk about this tomorrow when we talk about fusion, we have said to the Commission that we would try to keep Part 53 so technology inclusive that it might address fusion facilities.

And obviously the high level safety function, retention of radionuclides, would be applicable to even a fusion facility whereas reactivity, as an example, or even post-operation heat generation is less a concern. I won't say it's not a concern, but it's less a concern than it is for a fission system.

So, that is part of the rationale for the setup as we have it, for not listing the specific functions other than the retention of radionuclides.

MEMBER MARCH-LEUBA: Hey, Bill, what are you talking about? This is Jose.

I'm reading this as a rule. And you just need an example of something that you might want to consider to do what?

I mean, you might just well describe the RFB, but what does it do?

MR. RECKLEY: Well, what it does is if you go back to that, again, the first principles slide. Any designer will have to identify how they are planning to retain the radionuclides.

As Charlie mentioned, they all, what they all have in common is there's a hazard. And that's the radionuclides.

MEMBER MARCH-LEUBA: Okay. But what you're saying is, Paragraph 53 to 30B is irrelevant. And it doesn't tell me anything.

You just have to do, it basically says the famous joke, when in doubt, refer to Paragraph A. So either say something or don't say something. But what you're saying doesn't say anything.

I'll leave you with that concept. I mean, rules have to be rules. I mean, they have to the law and have to be well-defined.

When I read them, I need to know how to follow it. I don't know how to follow this.

MR. RECKLEY: Well it's, again, it's intended to give the designer enough flexibility to say, to retain radionuclides, what functions do I need. And so --

MEMBER MARCH-LEUBA: And that is the function of a regulatory guide, not of the rule.

MR. RECKLEY: And we would expect that there will be guidance in this area. And one way, well, their actually already is guidance in this area, in terms of Reg Guide 1.233 on the LMP, goes through an exercise of identifying those safety functions. Of laying out what would be needed in order to satisfy Paragraph A on the retention of radionuclides.

MEMBER MARCH-LEUBA: You know what, if I ask you, how do I satisfy Paragraph B, what do I have to do?

Paragraph B is something I must do because it's in the rule. What do I have to do to satisfy it?

MR. RECKLEY: Well, when we get into Subpart H, the content to applications, part of it will be to say, what functions have you identified in order to satisfy 53.230. And that would be, for example, somewhat paralleled with the requirement we currently have for light water reactors to address the GDC and for non-light water reactors to define their principle design criteria.

Those same things, as we talked about before are, the identification of the safety functions is part of that exercise.

MEMBER MARCH-LEUBA: Yes, I think I made my point clear in that --

MR. RECKLEY: Okay.

MEMBER MARCH-LEUBA: -- this babble makes no sense whatsoever.

MEMBER REMPE: So, I guess I have a different perception, but make sure I understand things correctly, Bill.

To me, they are going to go through. And if they have a unique non-LWR, they will look at things that could lead to radioactive, to release of radioactive materials. And if they, for some reason, if they have a chemical interaction and you have an air ingress or a water ingress in a gas reactor that can lead to radioactive material release, then that is a safety function that will be identified as an additional safety function that must be done.

And then I look at C, and even though some things are primary and some things are additional, all of those things have to be met. And so, the fact that it's a primary or an additional one doesn't mean the regulatory is going to say, oh, it's only an additional one. It gets the same attention as a primary.

Am I understanding the intent of what the words are here, Bill?

MR. RECKLEY: Yes. I think you probably worded it better than I did, so thank you.

And the reason that it's constructed the way we constructed it was, because it's to be technology inclusive, how you do B might differ.

And again, I don't like to use it too much, but ultimately if Part 53 is used for fission energy systems, it will have a different set of safety functions than fission reactors. Some of them will be similar, but they'll be different than fission reactors have.

But even within different designs, the importance of something like chemical interactions might differ. And so, anyway, I understand what people are saying, we'll just take that as an observation.

23. So licensing basis events are, again, I don't think the concept of having them has been very controversial. There has been some discussion on where this should be within Part 53.

But in general, what we're trying to emphasis is that any designer needs to look at a range of unplanned events from anticipated operational occurrences down to very unlikely sequences. And within LMP, if you want to go over to that terminology, from AAOs to design basis events and the lowest frequency events down into beyond design basis event category.

So, going on then to Slide 24.

MEMBER REMPE: Oh, one more thing, Bill.

MR. RECKLEY: Oh.

MEMBER REMPE: Actually, back to 22. The other thing I guess one I would raise to maybe address some questions would be that, if an applicant didn't identify chemical interactions and the staff reviewed it and said you need to look at this because you could have had a release, they would have to add that safety function as part of the review process for Part 53, right?

MR. RECKLEY: We would certainly be

raising questions. And quite possibly, it could lead to them adding that as a safety function.

MEMBER REMPE: Again --

MR. RECKLEY: Yes.

MEMBER REMPE: -- that's I think something that's important that might help with some of the confusion about this. But again, I came from a history and a prior career with an advance reactor component and that's what they were concerned about.

But anyway, go ahead. Thank you. I'm sorry to interrupt again.

MR. RECKLEY: Oh, no problem. Thank you.

MEMBER BLEY: Bill?

MR. RECKLEY: Yes.

MEMBER BLEY: I'm going to interrupt. We're about halfway through your slides and I think the next five or six are kind of really important and things we didn't talk about in depth at the last meeting. So I'm going to declare a break right now and then we'll come back and finish those.

So Part C will probably go a little faster than those. So, at this time I'm going to declare a break. And we will recess until a quarter till the hour.

MR. RECKLEY: Okay, thank you.

MEMBER BLEY: We're in recess.

(Whereupon, the above-entitled matter went off the record at 3:27 a.m. and resumed at 3:45 a.m.)

MEMBER BLEY: At this time we will continue with Bill Reckley's presentation. Bill?

MR. RECKLEY: Thank you, Dennis. So, one of the things we wanted to talk about is the licensing basis events.

There was some discussion at the subcommittee meeting and some distinctions of how it's done under Part 53 and the basis that we're getting out of the licensing modernization project and maybe how it was done traditionally, so the next few slides are kind of a summary or a revisiting of the LMP and the discussions we had with the ACRS during the development of Reg Guide 1.233, SECY paper 19-0117.

I don't know the protocol, Dennis, so I'll just offer up that I know there's new members. I also know this is one of those topics that if you're not exercising it, it's hard to keep in the forefront of your mind.

So, if for new members or as kind of a refresher for anybody, if there's a mechanism for us to give presentations or whatever informal processes, we're certainly willing to do that if there's an interest, and we can set that up if Derek or somebody wants to just give me a call.

We have existing presentations we're using for staff and in interactions with other regulators like CNSC for example, so we have all of that on hand. It's not really a burden for us to do.

MEMBER BLEY: Thank you very much. I think that's something -- we'll talk about it.

MR. RECKLEY: Okay.

MEMBER BLEY: It's something I think Paul might want to take advantage of, so go ahead, and this is one more of those areas where the real language just says we'll select them from this group, but it doesn't really go into what was in your SECY, what's in the LMP on exactly how you do that.

MR. RECKLEY: Right, so, but before, if we go on then to slide 24, before getting into the LMP, we just might want to revisit some of the ways it's been done in the past.

And so I know it's a busy slide and most of you are probably aware of the traditional approaches. I just copied this out of the standard ANS 51.1, the 1983 version.

Actually, for most operating reactors, it was the previous version, the one that's kind of highlighted that talks about condition two, three, and four events, basically the anticipated operational occurrences and design basis accidents. This is also discussed a little further in chapter, or, yeah, chapter 15 of the standard review plan under section 15.0.

But basically it just lays out, and it's similar for boiling water reactors and pressurized water reactors, and kind of follows roughly a process hazards kind of approach.

Consider what could make temperatures go up and down. Consider what might make flow rates go up or down. Consider what might make reactivity go up or down, what might disturb the power distribution within the core, what might lead to losses of inventory.

And laid out basically in the earlier versions was largely based on engineering judgment to define the categories in terms of anticipated events or events that were not considered likely to happen or conditioned for the design basis accident conditions.

The 1983, by the development of the 1983 standard, you can actually see that frequencies were being considered more specifically in both the categorization and also the little box is basically a frequency consequence curve.

So, by the 1980s, the earlier versions were starting to introduce or including more of a frequency component. Now, very -- I don't think any reactor was actually referencing the '83 standard. By that time, we weren't licensing plants anymore.

So, that kind of just lays out the background for the light water reactors. If we go onto slide 25, I'll go through a few slides that basically gives a similar process as it was developed under LMP, one difference being instead of using the -- it may be process hazard oriented terminology of consider what can make flow rates go up and down.

It basically is actually looking at event sequences from the PRA and looking at a particular component and failing it one way or another. In the end, it's similar.

So, depending on how you want to approach it, you can either highlight the differences or you can actually highlight the similarities between even the historical approach and the LMP.

MEMBER BLEY: Bill?

MR. RECKLEY: Yes, Dennis?

MEMBER BLEY: I actually had two things. The first is I agree with what you just said, but the underlying PRA --

(Telephonic interference.)

MEMBER REMPE: Dennis, I think we lost you.

MR. RECKLEY: Okay.

CHAIR SUNSERI: Yeah, Dennis, this is Matt. If you -- we can't hear you if you're talking.

MR. RECKLEY: Okay, what I might do, and Dennis has had some problems, I know, from the subcommittee meeting, so maybe I'll go on, and then when he comes back, we can pick up. Is that okay?

(Simultaneous speaking.)

CHAIR SUNSERI: He's asked me to carry on if he --

MR. RECKLEY: Okay.

CHAIR SUNSERI: -- drops off, so go ahead, Bill, carry on.

MR. RECKLEY: Okay, and we can revisit when he reconnects with the point that Dr. Bley was going to make.

So, basically for the LMP, the event selection is again taking the event sequences from the results of the probabilistic risk assessment and plotting them in terms of frequency and consequence onto this figure, and then as we get into more of the discussion, ultimately looking to the margins that exist between those events and the target figure, which is the figure in blue.

And if we go down to slide 26, this is a slide -- again, I'm using the MHTGR, trying to use it so we can maintain some consistency between what we're talking about, but this is one of the tabletop exercises done for the LMP.

It was actually done for X-energy XE-100 design, but where they were at this time. This was four years ago. They were looking and largely borrowed by the MHTGR PRA and event assessments, and you can see in purple all of the event sequences that they had identified in the various categories.

And then as we get into more of the discussion, actually the red dots, if you can see them in the design basis event region, are event sequences that contribute to the identification of a design basis accident, and we'll get into that discussion when we talk about the DBEs.

So, it does just show the number of events and the number of sequences that are being looked at from the probabilistic risk assessments, and actually even this, the number, if you plotted all of the sequences that were actually run, there would even be more dots on here.

They do group them into what's called event families based on similarities of how the progression and the end state of the transient, so this is actually a plot of families, not necessarily individual sequences.

But, so if we go on then to slide 27, this just lays out again the categories of events. Anticipated operational occurrences are basically those that go down to a frequency per plant year of ten to the minus two, where a plant year is any number of modules that might be affected, any number of inventories that might be affected within a plant.

DBEs are between ten to the minus two and ten to the minus four, and beyond design basis events below ten to the minus four down to five times ten to the minus seven, and then importantly, the methodology includes the assessment of uncertainties and the requirement to really look at that, and if the 95th or the fifth percentile in an uncertainty assessment puts you across the band, then you look at it in both categories.

So, the other thing that I'll just point out is, and this, we've introduced some confusion. In an attempt to not use exactly the terminology that's used in the LMP or to give the impression that we were requiring they use LMP, we're introducing different terms, but the terms have the same meaning, like instead of beyond design basis events, you might have noticed we call them very unlikely events, so, but within the overall construct, they're the same.

One last point on this, and I know I'm going pretty quickly through what can take many minutes to discuss, the other thing that's looked at under this is an assessment when you're looking at all of these event sequences against the cumulative risk metric, and again, that is proposed to be the QHOs in our particular example.

So, the other aspect, if we go to slide 28 --

MEMBER MARCH-LEUBA: Wait a minute, Bill, go back, go back. I wanted to make a comment on the record. How did you, I mean, how do you address in this methodology the known unknowns, which is what they call the completeness of the PRA? How do you know you selected all of the events at the front of that, not just the ones you thought about, but all of them?

MR. RECKLEY: Well, I think I'll let Marty get into that discussion a little more, but it basically goes through the methodologies that you use to make sure you address everything that can break.

MEMBER MARCH-LEUBA: No, that goes against the scientific method. You cannot prove a negative. You cannot say I -- you can say of everything I looked, this is how it turns out to be, but there might be something else I didn't understand.

And with light water, large light water reactors, we have 60 years of experience. Basically, almost everything that could happen has already happened.

With these large reactors, we don't have any experience and we have designers that want to expedite things. They don't, you know, have as much money. They cannot spend 20 years designing a reactor. They have to do it in two.

The completeness of the set of events is crucial. It's crucial, and as I keep telling you, I mean, this is not a hypothetical, okay? You can forget the most limiting events simply because it doesn't fit in what happened for the last 60 years on light water reactors.

And I don't see any emphasis on the rule or in your thinking on the review of the staff to understand how complete is that set of events.

MR. RECKLEY: Okay, let's --

MEMBER MARCH-LEUBA: And that's a clear -- I mean, you say you're premising the whole Part 53 on the fact that I can calculate the risk, and that's a non-scientific statement, period, over and out.

MR. RECKLEY: Okay.

CHAIR SUNSERI: Bill, this is Matt. I just want to let you know Dennis is back on, so.

MR. RECKLEY: Okay, let's revisit that point when we get to the PRA discussion, and I think Marty will, I think either in his slides or in the backup slides, go through the methodologies, but let me defer that. And Dennis, if you had a point before you dropped off?

MEMBER BLEY: It's an important point Jose raises, but it's not a point about PRA. It's about safety analysis. It would apply whether we were doing PRA or the other kinds of events, and the things that tend to dominate risks aren't things that we would have seen in 60 years.

They are things that -- we haven't seen everything. We're going to see some more things. And so I'll be happy when Marty gets to this and talks about it, but it's not strictly a PRA issue at all.

MR. RECKLEY: Okay, so, yeah, we'll

revisit this. Let me go on then to slide 29, 28, thank you.

So, one of the things that comes out of the process as we talked about are what are the required safety functions, things like heat removal and reactivity, and for those, those are the functions that have the potential to make you exceed the frequency consequence targets, and in particular in our example for the first tier safety criteria, the potential to exceed the 25 rem reference values.

That is what then goes into the determination of safety-related equipment because for every required safety function, you're required to have safety-related equipment in order to do what's on the next slide, the design basis accident, and demonstrate that using only safety-related equipment, you don't exceed the referenced values.

So, if we go onto slide 29, again, just coming back to the MHTGR example, they've done this exercise. They've identified the required safety functions, and then you go down into slide 30.

They would use only safety-related equipment to perform those functions in the DBA. So, they're going to have a safety-related reactivity control system, a safety-related heat removal system.

Many of the non-light water reactors are using reactor cavity cooling or reactor vessel direct cooling systems.

But in any case, you have a safety-related system for those required safety functions of needing to bring down the heat generation through reactivity control and to remove that heat through a decay heat removal system, so the DBAs are derived from the PRA sequences and then are looked at again only using safety-related equipment.

So, this is the LMP approach. It's also the approach that's reflected in Part 53 to have both a PRA, or as we'll talk about, another systematic assessment, and to keep a fairly deterministic DBA, traditional safety-related equipment as a kind of a backstop for the plants.

So, if we go down then to slide 31, the other couple sections that remained in Subpart B, the safety criteria, is the defense in depth 53 250. Again, we didn't make any major changes in the second iteration.

One change we did make was to emphasize that it's an engineered design feature, trying to give some room. If there's actually an inherent characteristic that's being credited, that would be given special consideration, but no single engineered design feature could be relied on to meet the safety criteria in 53 220(b), which again is meeting the QHOs.

MEMBER BLEY: Bill?

MR. RECKLEY: Yes?

MEMBER BLEY: That's a nice distinction, but it implies if you have an inherent feature, you'd be happy with a single one, and depending on what you -- they can be degraded as well, so it seems odd to suggest engineered design features to me.

MR. RECKLEY: And we're going to have to define some of these terms. Engineered design feature would include a passive system, so those can be degraded. What we're tend --

I mean, we're still developing this and engaging stakeholders on the terminology, but when we use the word inherent, it is something that doesn't require something even like natural circulation. So, it's not --

MEMBER BLEY: Even if it's coming from the physics, which I think is what you're saying.

MR. RECKLEY: Right.

MEMBER BLEY: You really got to be careful and make sure, one, you know all of the physics that might apply, and two, that nothing outside can interfere with it, so.

MR. RECKLEY: I 100 percent agree and that's -- and we're not trying to say it would be easy. So, if you have an inherent feature, you're right.

What we mean by that is it's the physics, but the physics has to be maintained over the life of the facility, so that means the physics couldn't be changed by irradiation or other environmental factors. It means the inherent feature is present within the bounds of what the plant's going to be operating under.

So, no easy task to show that the inherent feature can be relied on. They're going to have to do the science, the testing, and all of that to demonstrate that that inherent feature could be relied on, so.

MEMBER BLEY: Then you have plenty of external things like fires, severe earthquakes.

MR. RECKLEY: Right, so --

(Simultaneous speaking.)

MEMBER BLEY: -- all of those things.

MR. RECKLEY: We agree 100 percent. We were just trying to give some room that if there is such an engineered, I mean, if there is such an inherent feature and it can be proven, then that would be a basis to at least evaluate not requiring additional defense in depth measures, but no easy task. I'd agree with you there.

MEMBER PETTI: Bill?

MR. RECKLEY: Mm-hmm?

MEMBER PETTI: Does this mean as written that if one wanted redundancy and backup, you could have one engineered system and one inherent system?

MR. RECKLEY: That would be one way to address the potential uncertainties with the inherent characteristic that Dennis just mentioned, yeah.

MEMBER PETTI: Right, I can see some cases, some inherent functions where you can back it up. I can see others that it's harder to back up, for instance, molten salt. The fission product retention in that salt, I don't know you'd get an engineered system there.

I can -- probably an engineered system to make sure the temperature doesn't get outside some bound that would invalidate its ability to hold fission products or something, but okay, thanks.

MEMBER KIRCHNER: Bill, this is Walt. Along Dennis and Dave's line of thinking, why do you have to modify it with engineered? Why not just no single design feature, whether it's inherent, or passive, or engineered, or -- it's a design feature, something maybe as simple as a negative temperature coefficient or neutron leakage for reactivity control.

But again, as you pointed out and Dennis did in his examples, those things can be affected throughout the life of the plant because of upset conditions and so on and so forth.

You know, like a fast reactor that depends on leakage, well, you might not have that performance characteristic under all conditions, et cetera. Why not just leave it at single design feature and not have to split hairs over whether it's engineered, passive, or inherent?

MR. RECKLEY: We were -- I mean, one of the reasons is you have to go back and see how this is actually being used to assess individual event sequences, right, individual events.

And so whereas what we're talking about up to this point, I tend to agree with everyone that when you're talking at an overall plant, that is actually the way it would most likely play out, but when you're looking at an event, at a particular event, we're just saying if it could be proven that that event could be addressed by an inherent feature, we were acknowledging it's a challenge, but we were trying to say that that sequence then would not need to be backed up with an additional design feature, or may not have to be.

So, again, it was one of the comments that we had gotten. Some of the designers had felt strongly they could justify the inherent features, and so as a compromise, this is what we're proposing, but I guess that's all the explanation I can give.

MS. VALLIERE: Hey, Bill? I might add just to jog the members' memories that when we presented on key guidance documents that need to be developed to support Part 53, you'll find I think in that list that guidance on inherent characteristics was one of the items identified as needing guidance to support Part 53.

MR. RECKLEY: Thank you, Nan. Okay, so if we can go onto slide 32, the last section within Subpart B on the overall objectives and safety criteria is the need to protect plant workers. We largely do this by referencing back to Part 20, and I don't think there was much controversy to that, at least in the discussions with the subcommittee.

Going on then to the next section and the next subpart under design and analysis -- well, maybe I'll stop there and say is there any questions or discussion on Subpart B?

MEMBER BLEY: Bill?

MR. RECKLEY: Yes?

MEMBER BLEY: I was trying to make a point before I lost the internet.

MR. RECKLEY: Yes.

MEMBER BLEY: I don't know if you heard me.

MR. RECKLEY: Only the very first fragments, so, yes, if you could just repeat the two points?

MEMBER BLEY: There were two things I wanted to mention. With respect to your slide number 24, which is kind of nice, but the first note is by the time you have this kind of layout, you'd have 30 years' experience working almost exclusively with expert judgment to dream about what are the things that could go wrong and how do we consider them, so this was a real evolution by the time you got here.

And the other, I think I was just talking about whether you're doing this traditional approach to define your errors and design basis accidents that you're going to analyze in the traditional way or whether you're looking for initiating events in some areas in the PRA, it's the same process.

You've got to find them before you know what they are and that's a place where this time it looks pretty coherent, but that was after 20 to 30 years of trying to describe what these things ought to be and it was unique to LWRs.

So, this idea that you need a way to look for these events, especially for new technologies where we haven't been working on them for decades, that's where the guidance for people is very sparse. There isn't --

(Telephonic interference.)

MEMBER BLEY: Please continue with your next set of slides.

MR. RECKLEY: Okay, thank you, Dennis, and I'll also mention that, you know, one of the ones with the least experience is molten salts, and there are a couple of reports out of Oak Ridge going through an exercise similar to ANS-53.1.

There's also a good EPRI report that was supported in part by DOE that talks about how to do a process hazard analysis for molten salt systems, which are similar.

Yeah, we don't have much experience maybe with molten salt reactors, but chemical systems have been used in process hazards analysis for a long time, so they use that exercise.

And then the EPRI report talks about, as Marty will go into the PRA discussion, also how to inform or to use the process hazards as a starting point for what ultimately would go into the PRA, but we'll talk about that a little more under Subpart C under the analysis.

So, yeah, if we go onto slide 34, again, the layout of Subpart C follows the chevrons we talked about earlier, the design criteria, the safety functions in Subpart B.

Then they progress down into Subpart C where the first section is on design features, and then the second section, second and third sections are how do you define the functional design criteria to meet the first tier.

That's the safety-related design basis accident tier, and then the second tier, which is the more risk-informed approach coming out of the PRA, the beyond design basis events and so forth, and then how you get down into some additional design requirements we'll talk about, and then really we want to spend some time talking about the role of the PRA under the analysis section.

So, going through the first couple of sections, 35, I'll turn it over to Marty. One of the things that we wanted to talk about because it had come up in the subcommittee meeting was single failure versus the PRA probabilistic and reliability approach, so, Marty?

MR. STUTZKE: Yeah, good afternoon. I'm Marty Stutzke, the senior technical advisor for probabilistic risk assessment in the division of advanced reactors and non-power production and utilization facilities.

And as Bill had said before, we wanted to talk about the fact that Part 53 would allow the single failure criteria to be replaced with a reliability criteria.

This had been mentioned in Reg Guide 1.233 as approved by the SRM, the SECY-190117, to allow us to do this, as well as using probabilistic evaluation to select events, some things like that.

A little bit prior to that in a different context was the staff had approached the Commission in SECY-19036 about the NuScale ECCF systems, specifically the inadvertent actuation block valves, and whether single failure criteria should apply to those valves specifically, and the Commission came back and told us to apply risk-informed principles when you don't need the strict deterministic criteria such as the single failure criteria.

MEMBER BLEY: Marty?

MR. STUTZKE: Yes?

MEMBER BLEY: This may be more for Bill. When you're all talking about the single failure criteria as applied to a system, and a big system that has a safety function, under the single failure criteria, it has to be able to withstand any single failure without a loss of function.

There's another aspect of single failure that Bill was talking about earlier, and that is when you do the equivalent of the Chapter 15 analysis deterministically with only safety grade equipment operating, you assume for each system the most challenging single failure.

That is still part of Part 53 as I understand Bill's earlier explanation. Is that correct, Bill?

MR. RECKLEY: Actually not. The defense in depth measures that we talked about would require that you have additional measures, but the difference between what we're proposing and the traditional single failure criteria is that we wouldn't require for the DBA a specific additional single failure that has led traditionally to two trains.

MEMBER MARCH-LEUBA: It would be perfectly okay with you to have a single safety protection system, a one channel protection system because you would have to assume a single failure?

MR. RECKLEY: One train, yeah, one train.

MEMBER MARCH-LEUBA: So, your I&C will not only not have diversity, it won't even have redundancy?

MR. RECKLEY: For the assessment of the DBA. The diversity --

(Simultaneous speaking.)

MR. RECKLEY: For the diversity, and the redundancy, and so forth comes largely in repeating that in a non-safety related system most likely for the other event sequences.

MEMBER MARCH-LEUBA: So, you will have one safety grade I&C channel and say three non-safety grade channels, trains? That would be perfectly okay?

MR. RECKLEY: I'll be honest. I haven't seen the application of this down to the I&C channel. I'm mechanical oriented, so --

MEMBER MARCH-LEUBA: Okay.

MR. RECKLEY: -- I've seen it on the mechanical --

MEMBER MARCH-LEUBA: You're a mechanical guy. Would it be okay to have one single safety relief valve to protect for the SME safety code?

MR. RECKLEY: For the DBA, yeah.

MEMBER MARCH-LEUBA: Yeah, so only one safety relief valve will be okay for you --

MR. RECKLEY: But keep in mind --

MEMBER MARCH-LEUBA: -- to protect this against other pressure?

MR. RECKLEY: For the DBA. Because you have to analyze the other events and meet the defense in depth requirement, you will have more than one.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: But it will not be safety related.

MR. RECKLEY: It may not be safety related.

MEMBER MARCH-LEUBA: So, only one safety-related SRV, only one safety-related protection channel, only one control, okay, that's fantastic, man. You're making my day.

MR. RECKLEY: Keep in mind that you're talking about what is needed to protect against an individual event sequence, not what would be found acceptable for the overall plant design, because you need to bring in the other event categories, and the defense in depth requirement, and other --

MEMBER MARCH-LEUBA: The only event categories, the ones that don't have AOOs don't give you safety-related components. Are you saying that we're going to create non-safety grade, some additional control protection system channels and trains, non-safety grade SRVs we're going to give them credit for? They're not in tech specs and do not exist, but we grade them? Okay, guys, you know how I feel about this thing. This is lunacy.

MR. RECKLEY: Okay --

MEMBER BLEY: Bill? This is -- yeah, when you do the DBA analysis, which Part 53 calls deterministic, that's fine and conservative, but you assume everything's working. You're not doing reliability accounting for the chance of failures. You're assuming everything works, so it's a different kind of analysis that we did before.

Now, I will agree with you if you've done the PRA right, you've looked at the overall risk and the chance that things fail, but if you do that and you come up with those licensing basis events which are out of the PRA, those are pretty reasonable, but then when you say I'm going to define a DBA as one of those and I'm going to analyze it in the traditional deterministic conservative way, you're not doing that.

Now, there are good arguments about why you might not want to define DBAs, just stay with the overall PRA, but I don't see what you gain at all by defining DBAs and then applying thermal hydraulics to it. I don't get it.

MR. RECKLEY: The notion is that you'll have, at least for the required safety functions, you'll have at least one safety-related way to meet that function. So, in reality, you have multiple, but at least one of those paths will include only safety-related equipment, so.

MEMBER BLEY: But if we go back to thinking about an LWR, when we have, say, three pumps of safety injection and you go buy one that's safety grade and two that aren't, you're probably going to buy the same pumps. I'm not sure what we're picking up here.

MEMBER REMPE: Well, would we say the maintenance might be less for the non-safety grade ones?

MEMBER BLEY: It might be nonexistent.

MEMBER REMPE: Right, so --

MEMBER BLEY: And it can't be, and you can't just have one because when you do that PRA with its embedded systems analysis, you cannot get sufficient enough reliability out of the system if you don't have maintenance and if you don't have redundancy in the systems.

You can't approach anything like returns of reliability we need in our systems to protect the design. I guess where I'm -- it sounds like the only thing doing this defining of the DBA that does anything is that the main one of them is safety grade. They're all going to have to be under tech specs or you can't get the maintenance contributions on reliability well enough.

MR. RECKLEY: And we'll get there when you see the operating controls we set. There will be reliability programs for -- let's take your example and there's three ways to remove heat.

Yes, the exercise is one of those ways, if it's serving a required safety function, it will be safety related. They will all have, if they're risk significant, they'll all have reliability controls on them.

What you'll see under proposed Subpart F is the safety related would have the control under tech specs and the other two would have their controls under a reliability assurance program, not in tech specs, but in another required by regulation program, which is the reliability assurance program, so.

MEMBER PETTI: So, Bill, does that -- I mean, I'm trying to understand does that change anything really on the ground that there's requirements, but they're coming through two different pathways if you will in terms of what you do with the systems on the ground?

MR. RECKLEY: The thought is there would not be that much difference on the ground. From a regulatory perspective, the tech specs will have the traditional action statements and so forth, whereas the others would come more under a licensee-defined program, so a little more flexibility in the non-safety related, non-tech spec.

In terms of the actual equipment like you're suggesting, probably not that much difference, but in the regulatory treatment, some difference.

MEMBER PETTI: Okay.

MR. RECKLEY: So, I'm sorry about --

MEMBER REMPE: When I think about like crud that was deposited on the vessel head nozzles, how often they have to be inspected could change, which could affect the performance, right?

And so I'm trying to think of examples with real operating plants where you had to do stuff and there was an inspector who was verifying it was done. I mean, yeah, it would save a lot of money for the licensee, but I'm wondering does that mean we're really depending on one?

MR. RECKLEY: Yes, some of this, I think, and I hate to say it, but I think it will be more clear when we look at the operations requirements, and that will be next month, so.

MEMBER KIRCHNER: Bill, this is Walt. I guess I'm -- going back to my colleague Jose's comments, let's just pick something, a reactor protection system.

To only have one channel, to me, violates the whole philosophy of defense in depth, one safety-grade channel for detection of, let's say, over power, you know, high flux calibrated in terms of power, so a power trip.

MR. RECKLEY: And if we could, Walt, just because I'm not as familiar with the I&C side and the fact that when you get into I&C, even in the safety-related functions, you've going to have multiple channels.

Those channels might actually all be safety-related channels because they're looking at different quadrants in the core. They're looking at different loops in the coolant system and so forth.

So, I prefer not to focus in on an I&C channel, but if we look at, let's say, a heat removal system and take reactor cavity cooling as the safety-related system, all we're saying is there wouldn't be necessarily two trains of reactor cavity cooling, but reactor cavity cooling is not the only heat removal system you have.

In fact, it might be the fourth or fifth heat removal system that you have, but it might turn out to be the safety-related system you have for heat removal.

MEMBER KIRCHNER: Well, let's take it on the mechanical side, just kind of rhetorically, your nice chart of the layered fission product or radionuclide barriers.

So, and maybe the first one is the equivalent of the, of a fuel form, or the first one is for a liquid fueled system is probably that primary envelope.

And if you lose that first, you know, that first line of defense, one would think that you would need a safety grade second line of defense or, yeah, safety related, I'm sorry, equipment, you know, qualified to be in concert with your defense-in-depth overlying not philosophy now but objectives in terms of --

MR. RECKLEY: Right.

MEMBER KIRCHNER: -- system performance.

MR. RECKLEY: And I think we're probably in agreement other than the safety classification of your backup.

The fact is you would be required to have a backup. But the backup, depending on the assessment that you're doing, the backup would very likely be a non-safety related backup.

And that's not dramatically different than what we accepted on some of the passive light water reactor designs. But it -- there still would be a backup. There still is defense-in-depth. You're not totally relying on one layer as you're suggesting.

It's just that, because of the way we've categorized the events, the design-basis accident, you're going to credit the safety related one. You're going to ignore the non-safety related ones that are actually providing that backup.

And that way you're assured, again, that you have at least one safety related way to carry that, to make sure that you don't exceed the 25 rem first tier safety criteria.

(Simultaneous speaking.)

MEMBER KIRCHNER: I understand what you're saying. I just, you know, I've said this too many times in the past. But I'll say it one more time.

I kind of look at this and say, well, does this provide an equivalent level of protection in the public's eye, I mean, because that's, you know, if that defense-in-depth, the second barrier now, is not safety related, do you convince the public that you've provided an adequate, a comparable level of safety to the existing fleet. And I don't know.

It strikes me that the public, looking at this not knowing the nuances of an in-depth PRA, et cetera, et cetera, might not be convinced.

MEMBER HALNON: Bill, do you ever see a situation where the backup would be non-safety related but important enough to be tested by a tech spec surveillance?

MR. RECKLEY: We're still developing the requirements under Subpart F. That question comes down really to the fourth criteria and under 50.36 for including limiting condition for operation and a tech spec surveillance for risk significant SSCs.

And since it's out, I'll tell you, our first draft, our first iteration of the language says that that's addressed through the reliability assurance programs for the non-safety related equipment and not included in tech specs. That's our first iteration.

MEMBER KIRCHNER: But what happens, Bill, when the PRA results are used such that that second backstop or second line of defense after the safety related equipment has been assumed to fail or does fail and it's not on the D-RAP?

And I don't want to go into actual details, but when we have instances where in the recent review the two obvious systems to recover or provide that backup didn't make the D-RAP list.

MR. RECKLEY: Well, again, well, the assumption -- and I'll let Marty get back into the slides here. But the -- if it's shown to be either risk significant because of the PRA results or it's required to meet the defense-in-depth measure, under what we would propose under Part 53, it would be in the equivalent of D-RAP. That's one of the criteria for being there.

MEMBER KIRCHNER: But let me pursue it a little further then. So, okay, we obviate the need for single failure criteria. We do define -- this morning we heard about fire safety. I'll use their terminology, a success path to fall below, in that DBA analysis, fall below the dose, the safety criteria as expressed in terms of dose at the exclusionary boundary or LPZ.

Wouldn't, in that case, wouldn't that second line of defense then have to be covered by the D-RAP, or as Greg was saying, Greg triggered me on this, that, wouldn't then that have to be somehow in the tech specs at least for the DBAs?

MR. RECKLEY: Again, and we're jumping ahead a month to look at what's in tech specs. But, yeah, if it's required to address the DBA under the Part 53 proposal we just released, then it's required to be in tech specs.

MEMBER KIRCHNER: Okay. All right. Thank you.

CHAIR SUNSERI: Hey, Bill, this is Matt. Vesna has her hand up. You might want to call on her.

MR. RECKLEY: Please, Vesna.

MEMBER DIMITRIJEVIC: Yes, hi. So my question is to Marty. Marty, are we discussing, because now I'm confused with all of this.

Are we discussing here, so action of the licensing basis events and application to Chapter 15, or we are discussing safety classification, because I have a question for later on safety classification, but suddenly we are discussing safety classification here. And I didn't see too much about safety classification in your documents.

So are we discussing here safety classification of SSCs, or we are discussing selection of licensing basis events? Those are two separated things.

I mean, you say in selection of licensing basis events, credit only safety equipment. But where is the safety classification and how it's determined that's not discussed.

MR. RECKLEY: Yeah, no, I might have gone through the slides quickly. If you can flip back up a couple to -- and the topics you mentioned are all interrelated.

If you go back to 27, so this is the selection of the licensing basis events by looking at the PRA sequences and putting them in these categories --

MEMBER DIMITRIJEVIC: Yeah.

MR. RECKLEY: -- based on the frequency. Then if you go to the next slide 28, you're looking at those and saying, out of those event sequences, if I take a piece of a -- what are the required safety functions?

So, if I don't have any system to remove heat, what would happen to the event sequence that I just plotted in the design-basis event category?

Let's say I have three ways to remove heat. And that's -- so that puts me in the design-basis event category, because probably at least two of the three is going to work on any given sequence. But if I take away all heat removal, what happens?

And that's what that arrow is showing. If I take away all heat removal, then I'm likely in this example to exceed the 25 rem number. I'm going to exceed that frequency consequence target figure.

That means that's a required safety function, because without it I won't pass the criteria, the first tier criteria.

So now that goes into safety classification. Given I have identified that as a required safety function, I need at least one system to perform, one safety related system to satisfy that function, to perform that function.

So one of, in the example, one of the three heat removal systems is going to get picked as a safety related system.

MEMBER DIMITRIJEVIC: Okay. So let's start with this selection. This selection of event sequences where they actually, the end state is not determined yet. So it could be an initiating event, everything successful, for example. That could be one of the sequences, you know.

So let's say this sequence, if you want to determine this lease in this category but you're only crediting safety equipment, so we have here a question of the chicken and egg.

I mean, how do you select sequences if you don't know what the safety equipment or, I mean, you know, you can see how this is all -- and somebody was proposing that maybe we go as an example to this, I think it was Bill, to this licensing basis event selection, you know.

But we can actually do this as a tabletop to see how that will work, because obviously these things are so interconnected. It's not clear at all how that will work in practice.

MR. RECKLEY: Right. And --

MEMBER DIMITRIJEVIC: It's not clear to

me.

MR. RECKLEY: Yeah, and there's been various tabletops done. And so we could revisit that, as I mentioned to Dennis, through the LMP briefing or whatever.

MEMBER DIMITRIJEVIC: I mean, examples in that Lorad (phonetic), you know, which has examples for PWR and PWR, the 1860, in those examples it's already known, because they rely on the existing light water reactor.

Like, for example, if you just look at heat removal, the main feedwater, and feed and bleed, they're non-safety functions. But here when you are playing with, and then they're only looking in, you know, absolutely feedwater removal drains.

But if you are looking in the new designs and you don't really know what is happening, then you don't know how to select those sequences in the basis. So this is a -- you know, like looking at examples will help a lot in these cases.

MR. RECKLEY: Okay. So, Marty --

MEMBER PETTI: So, Bill, just, let me make sure I understand, because this is kind of moved around.

Let me take an example that I know and you know, MHTGR, the regular cooling system was one way to remove heat. It's clearly safety related. Reactor cavity cooling system was a passive system. And it was also safety related.

But there was a third cooling system that many people don't know about called the shutdown cooling system. It was not safety related. And so it backs up, if you will, the passive system.

And it was decided by the designer. The designer could have decided the shutdown system could be the safety system and the RCCS, reactor cavity system, could have been non-safety. But they made the decision to go the other way.

So there were three ways to remove heat, two, you know, one engineered and safety, one passive and safety, and one engineered but non-safety. And that would be consistent with how you described, you know, single failure of a system.

MR. RECKLEY: Right, right.

MEMBER PETTI: Okay.

MEMBER KIRCHNER: But in your case, which I know well, Dave, there were two safety grade systems for the function. And as you said, they made the decision.

It was a tradeoff between, you know, the cavity, passive system versus the active shutdown system that would get them down, you know, to low enough temperatures for refueling. But there were two safety related systems that could take the heat out of a core.

MEMBER PETTI: Right. Yes, yes. But I think that, the defense-in-depth requirement says no single individual system. That's what the words say. So this idea of having only one system would fail, right --

MEMBER KIRCHNER: No, I --

MEMBER PETTI: -- from that requirement --

MEMBER KIRCHNER: -- Bill earlier correctly that second or third system might not be safety related.

MR. RECKLEY: Right. And you also, even had to be careful in the example, because the pressure boundary might have been safety related for another purpose other than heat removal. So --

MEMBER MARCH-LEUBA: Yeah, on all the defense-in-depth you're hanging your fruit on, your hat on, it is so full of examples and such as and maybe you can use margin and maybe you can use hand waving. It's not clear that defense-in-depth says anything honestly.

If you want to take credit for defense-in-depth for your backup, your safety grade systems, you have to tighten up the language and make sure that defense-in-depth truly exists, because right now you have so many qualifiers and examples and how you can do this, how you could do that, that I don't have any good feeling that it exists. Thank you for listening to my complaints.

MR. RECKLEY: Okay. Back to 35, Marty.

MR. STUTZKE: Yeah, on the last row, I would just point out that this notion of replacing the single failure criteria with the reliability criterion has been around about 18 years. And the Commission approved it back in 2003.

Slide 36, please, a little history on single failure criteria I thought that would provide some perspective.

Back in 1965, the Atomic Energy Commission convened a regulatory review panel to look at ways to review policies and practices for licensing with an eye towards expediting the licensing process.

And the panel came back. And one of the recommendations was they felt there was an absence of definitive requirements and criteria. And so to that end, the Atomic Energy Commission proposed the general design criteria in late 1965. They weren't finalized until 1971.

But later on in '77, as the slide points out, the Commission asked the staff to critique the use of the single failure criteria. And the staff said, yeah, it seems to be working, however, it's just one of multiple fuels that are applied in system design and analysis with the not comment, the single failure criteria in and of itself is not sufficient.

They also pointed out the single failure criteria was developed without testaments of probability, some components or system failures.

Most importantly, they picked up on the insights from WASH-1400, the original nuclear plant PRA, and said things such as systems interactions, what we would now call dependent failure analysis, multiple human errors, tests and maintenance, all of these things have an influence on reliability. And at the time, they're not considered within the scope of the single failure criteria, so we have to use additional methods.

And one thing that I found very interesting, almost prophetic, it says, gee, the use of probabilistic methods such as the reactor safety study, could be, areas could be increased and ultimately supplant the single failure criteria.

So, when you think about it, that was some-40, well over 40 years ago. Okay.

MEMBER BLEY: I'd like to fill a little bit more in on your history there, Marty. Back when they did the reactor safety study, most folks here weren't -- you know, they hired a bunch of guys from Boeing to come over and bring folks and analysis with them.

And those guys have (audio interference). But then when we analyze these systems, you're going to find that even more single point failures than we would have ever guessed and reliability is much lower than we expected.

That turned out not to be true. And it turned out not to be true because of these single failure criteria and the way the staff at that time, you know, I've brought in some system analyses to talk with the old generation of staff on what, dependent failures.

They did track down some of these repeated interrelated, interacting system failures and really developed a deep questioning to look for single failures. And that served very well.

You know, the folks found very few of

those failure in likely failures in systems. So it has worked well. But it turns out that it missed some of the more important things that were identified as important in the study.

Anyway, I'm sorry, Marty. Go ahead.

MR. STUTZKE: No, you said it very well. That was all that I was going to comment on this slide. So slide 37, I think this one is yours, Bill.

MR. RECKLEY: Okay. One of the other questions that came up during the subcommittee was codes and standards and the phrase generally accepted codes and standards.

So, since that time, we have released some definitions. And one of the definitions we released was of consensus codes and standards, which I won't read here.

But basically it is our general understanding. It's coming out of a standards development organization and run through the normal, you know, processes of ASME, ANS, ANSI, so forth.

So the, in terms of the discussion box down below, we wanted to continue to encourage and actually are required to encourage the use of consensus codes and standards. So that's one of the reasons we put the language in, to satisfy the

National Technology Transfer and Advancement Act.

We do, however, recognize that there's a lot of possible technologies in play and a lot of different potential standards and somewhat of an interest to also look at other standards approaches like the International Standards Organization, or ISO, and their standards in some areas for some components, as well as the possibility of looking at other international standards, if it happens to be a vendor or a designer that's looking let's say to deploy in Europe first, or some other area where another set of standards other than, for example, ASME or IEEE might be the ones generally used.

So, given the whole host of potential standards, that was another reason we stuck with wanting to encourage the use of consensus codes and standards but not incorporate into the rules specific codes and standards like the boiler and pressure vessel code that we have for light water reactors.

We would look to, I think as we discussed during the subcommittee meeting, look at guidance documents, the submittal of proposals from either SDOs, which we currently do, or the individual designers or others and to try to pick that up in guidance.

And a recent example of that, for example, is the Division 5 of ASME for high temperature materials that we're looking to pick up in a Reg Guide but not necessarily incorporate into 50.55(a).

And the last bullet there, one of the reasons for that is that the incorporation of those consensus codes and standards into the regulations has raised other issues, including the need to do rulemakings when they come up with new versions of the codes. And that would be a little easier to handle in guidance updates versus rulemakings.

So that was the slide on consensus codes and standards and why you're not seeing ASME or IEEE or ANS standards incorporated into Part 53, at least where we are with the preliminary language.

So, if we go on into 38 --

MEMBER HALNON: Okay. Bill, just --

MR. RECKLEY: Yeah.

MEMBER HALNON: -- this is Greg. Just one point on the consensus standards, in the guidance, you know, I think it's a good idea, because it is poised to get it into the rule and it takes a long time.

But do you foresee possibly taking any kind of major exceptions to portions of the code? I mean, that concerns me a little bit where the Reg

Guides come out and they'll have exceptions and/or, you know, a differing thought process on a certain endorsement. And you say we'll endorse, you know, nine-tenths of it but not this last tenth of it. And that could circumvent the use of it, the way it was intended to be used.

MR. RECKLEY: It can. We always reserve the right to do what you're saying, to put in exceptions or clarifications.

Generally, we're able to avoid that in many cases, and keeping in mind that often NRC people are on the consensus code and standards, so we can at least recognize what's coming and sometimes even influence what's in the standard itself.

But, so hopefully -- I agree with you. Hopefully, we could avoid that. And we traditionally have avoided it in large part. But we do need to maintain the ability in the development of a Reg Guide to take exceptions to anything in a consensus code and standard.

MEMBER HALNON: Okay. I guess it can be important to point out that all that would go through public comment, in addition to probably ACRS review as well for --

MR. RECKLEY: Right, right. Yeah.

MEMBER HALNON: Yeah. So, okay. Thanks.

MR. RECKLEY: Yes. All right. Okay. 38, on the PRA, again, the requirement is being maintained in our second iteration of the language to require PRA to be done. And the use of the PRA as is highlighted there is to at least support the assessment against the second tier safety criteria of meeting the QHOs.

And with that, I'll turn it over. I think the next slide --

MEMBER BLEY: Bill, I wanted to go back to the point Greg made. I wanted to support the staff in this area, because I have not seen a case that I was involved in where the NRC was considering adopting a consensus code or standard in which the NRC didn't have one or more people on the committee that was developing the standard.

So they were very knowledgeable about how it was developed and what the intent was. So I don't think there's much chance that you, you know, lose the intent.

Usually, the clarifications and exceptions are cases where the standard wasn't strict enough for what NRC thought was the appropriate --

MR. RECKLEY: Okay. Thank you, Dennis. I do want to caveat that when you serve on a standards committee you're not representing the NRC. I mean, there's the practical thing that you work for the NRC and you're volunteering to be on a standard. But you don't represent the NRC. I think most people know that. I just think it's worth --

MEMBER BLEY: That's a good point. So, when you come back to the NRC, you can represent --

MR. RECKLEY: Yes. You don't forget what you just sat through. That's exactly right. That's the point.

So, with this, I think one of the things that came out of this subcommittee was a need for a bit more discussion on PRA. So I'm going to hand it back over to Marty for slide 39.

MR. STUTZKE: Okay. This slide looks at past and present uses of PRA. These are listed in Standard Review Plan, Chapter 19, which typically applies to LWRs. But in general, these uses would also apply to non-LWRs.

So the first one, it's about identifying severe accident vulnerabilities. That one comes from the advanced reactor policy statement, which in turn references the severe accident policy statement.

The second one is the demonstration that the plant needs to commission safety goals. This bullet pointed out our second tier criteria would embed the QHOs directly into the rule language. This one comes, again, from the Commission's advanced reactor policy statement.

A third one here is use of the PRA to support environmental reviews, specifically the evaluation of SAMDAs, severe accident mitigation design alternatives.

Now, to be clear, Part 51 does not require the use of a PRA, but this is the way that it's been done in the past. And I refer you to Regulatory Guide 4.2 in general on the preparation of environmental reports and this new interim staff guidance 29, which talks about environmental reviews and SAMDAs with respect to micro-reactors.

But we point out in order to implement the methods in these things you require a full level 3 PRA. Because it's a consequence, the idea is to compute consequences of accidents, monetize them, and then compare to the cost of implementing the corrective action. So, in that respect, it's similar to a --

(Off mic comments.)

MR. STUTZKE: I would also point out that all the plants that have been certified designs and combined licenses all have level 3 PRAs in order to meet this environmental review and the use of SAMDAs.

So, in the fourth bullet, if you're going to implement the LMP guidance in NEI 18-04 you're using the PRA to select and classify SSCs, and inform defense-in-depth evaluations.

Let's go to slide 40.

(Off mic comments.)

MR. STUTZKE: Request people to mute their microphones, please.

Okay. On slide 40, for applications that are not based on the LMP, a PRA could be used to support the Ritnis (phonetic), the identification of systems incorporated within the program, et cetera. The results and insights to the PRA are used to support ITAACs, tech specs, COL action items, and things like this.

Of course, the PRA may be used also to support other concurrent voluntary risk informed applications that may be included within a license application, for example, risk informed in-service inspection, risk informed tech specs. All of these things could be in there.

And lastly, the staff uses the results of the PRA to inform the scope of the review. This was an action that came from former Chairman Jaczko and Commissioner Apostolakis. And it's known as the enhanced safety focus review approach that's explained better in, again, in the SRP like this.

But the idea is to focus the staff's review on what's important and do a smaller amount of review for things that the PRA says are not important.

Last and not least, the results of the PRA are used to support reactor oversight programs.

So, continuing with slide 41 --

MEMBER KIRCHNER: Could you go back to 39, Marty? This is Walt Kirchner.

MR. STUTZKE: Yes, of course.

MEMBER KIRCHNER: You know, I just -- this is just an observation from a non-practitioner of PRA but one who appreciates it.

I think the most important use of a PRA is to gain insights and inform the design. And that rarely ever gets listed.

It seems like the PRA is being used more to determine regulatory compliance, to exclude things from being on a D-RAP list, to, et cetera, et cetera, which are all I think useful and important things. But the most fundamental thing in my mind for the PRA is to use the insights you gain to improve the design.

And I just make that as a comment, that I would hope that, to the extent that the rule is requiring the PRA and such, that that is embedded in the, one of the purposes, rather than I think there's a tendency to, from my observing things on the committee over the last five years, to focus on the numbers and then use those numbers to exclude things from regulatory treatment or, et cetera, et cetera.

And that's all justifiable. There's economic reasons behind that.

But, again, I feel the most valuable part of a PRA will be at the design phase to help inform the design. And that rarely is cited.

MEMBER BLEY: I should, I really disagree with your last statement. I agree with, indefensible. But at least 4 of the design certs we've done in the last 15 years made heavy use of their PRA in the design process. In fact, it's what led them to the new designs they proposed and got in.

(Simultaneous speaking.)

MEMBER KIRCHNER: I stand corrected.

MR. STUTZKE: Absolutely. So we expect in the, is that the Commission argues in its advanced reactor policy statement that it's clear the Commission's intent was for designers to use the PRA as part --

MEMBER KIRCHNER: That's where I agree. That's I guess what I was getting at. And I stand corrected by Dennis.

But, yeah, when I think of the 2008 advanced reactor policy statement, it's just some of those concepts, if they were -- I think many of them, of those are embedded already in your language that you've been developing.

But that one just doesn't stand out to me. Maybe I'm missing it somewhere or maybe, as Dennis says, it's just done and that's it. But --

MR. STUTZKE: We'll take it under advisement.

Okay. Another thing that I wanted to discuss here is, in the letter the ACRS wrote on our Part 53 white paper back in September of last year, 2020, you all used the phrase it's important to be, to search for events without preconceived expectations.

And I know the topic had come up before about how do you know that you're complete. So I wanted to provide you with some language or some thoughts that come out of the non-light water reactor PRA standard, the various requirements on how the initiating events are selected and how one confirms or attempts to show that they're complete like this.

So there are requirements to identify initiating events, which are defined as challenges to plant operations, and mitigate those challenges such that you prevent a radioactive release.

That's put in there to account for things like you may have, for example, a loss of feedwater, followed by failure to scram. So the feedwater event would become the initiating event, and the scram failure and ATLAS sequence is treated elsewhere in the PRA.

The second requirements are using a structured systematic process. And it specifically lists things like master logic diagrams, heat balance fault trees, a process hazard analysis, failure modes and effects analysis.

The process hazards analysis, the PHA, has been a subject of study by the Electric Power and Research Institute and its contractor, Vanderbilt University. And they have actually applied it to the old molten salt reactor experiment design to use it as kind of -- I think of it as the prelude to the PRA, so a very good process. They've issued reports on this. So the guidance is there.

I should back up a minute and remind you that the PRA standard specifies what to do. But it doesn't tell you how to do it. Rather, it makes references to other techniques that could be used like this.

So, down to the third bullet, analyzing operating procedures and practices to see where humans could become involved and inadvertently trip the plant off the line.

The fourth bullet is still in the standard, review existing list of known initiators specific to type. Obviously, that bullet by itself is not sufficient.

One could come and say, take a list of LWR initiators and say, gee, I'm designing a molten salt reactor, so that one doesn't apply, that one doesn't apply. And you don't -- ultimately you end up with very few initiators.

So it's the totality of all these requirements on this slide and the next one is what provides the confidence.

That being said, conferring or referring to known list of initiating events is an appropriate way to do it.

MEMBER BLEY: Marty?

MR. STUTZKE: Yes.

MEMBER BLEY: Comments. This is very good. To my knowledge, the non-light water reactor care is standard.

Has it been adopted as yet?

MR. STUTZKE: We are in the process of endorsing that in a regulatory guide that will look very similar to Reg Guide 1.200.

MEMBER BLEY: Yeah. And we'll see that sometime. But, it isn't there yet.

MR. STUTZKE: Right.

MEMBER BLEY: The take we were trying to make is that, of course, you should look at existing lists, your last four.

But, really, that should be the last thing you do. That should be a check on was there anything in your other processes that you found out earlier?

If you're stuck with that list, it gets harder and harder to really dig in for these other approaches to try to make sure you're complete.

Anyway, but personally, I agree it belongs on the list. But, I think it belongs at the end after you've done the creative work of working hard for -- and using the things that might be hiding in your design.

And I'll be glad when this standard is

out. But, this is the first time to my knowledge. It's not the first time people have done this, it's the first time to my knowledge it's been on any kind of documents.

MR. STUTZKE: Yeah. That standard we intend to go to start the Reg Guide publication process towards the end of June or early July.

I know we have a meeting set up with one of the subcommittees of ACRS to talk about it. But, we hope to issue that standard by December, or endorse that standard by December of this year.

MEMBER BLEY: That's good news.

MR. STUTZKE: On your last comment about you were referring to the known initiators. But personally, I've always looked at that.

That's like when you do, you know, a calculus problem in school, and you know the answer is in the back of the book.

So, you do all of the creative work up front, and then you look in the back of the book and see if you got it right. Something extra that maybe you should have thought about.

MEMBER BLEY: Well, it certainly is. But, I'll give you one, an anecdote, and this comes from a lot of research in the area of expert elicitation.

And there's been a number of studies where they looked at the problem of anchoring, which is the problem if you start with the last bullet. And with a couple of the studies, they got people together and they said, you know, just to help you out, we're going to make up a first starting point.

And then you think about what could make it more like or less likely. You know, you work from that point.

But, now it's just made up. And if you start with that, it's amazing how close you stay to it by the time you've done the process.

You really don't want to bias yourself to some anchor point where you've been searching broadly.

MR. STUTZKE: All right.

MEMBER BIER: Dennis, this is Vicki. If I can just expand on that.

There was one study that specifically did this for fault trees. Where they had like auto mechanics or something.

And some of them looked at a complete fault tree for why a car might fail to start. And some of them looked at a fault tree when the cap causes were missing.

And both trees were rated as being equally completed. So.

MEMBER BLEY: That's their problem. Okay, thanks.

MR. STUTZKE: Okay. Let's go to slide 42, please. Okay. So, in addition to the previous slide, the list continues.

It says, you know, don't forget about the external hazards. Including combinations of hazards like seismically induced fires.

Looking at operating experience from similar plants if it's available. Basically a systematic evaluation down to the subsystem of the train level.

Including all of the supporting systems. So, you really understand the dependencies that the gear system has with other systems, and things like that.

Including initiating events that may have involved multiple failures if they arise from a common cause. That picks up things like earthquakes, these big ones like this.

Interviewing plant designers and operators after you've done your homework above like this. And last but not least, don't forget to consider initiating events that might impact multiple sources of radioactive material.

The non-LWR PRA standard would consider that you could have multiple reactors onsite, plus non-reactor radiological sources.

So, spent fuel or off cast systems, things like that. And they would all be included in the scope of the PRA.

So, okay. Moving to slide 43. Another question that we commonly have to address is, what about the lack of operating experience?

So what I've tried to list here in the lefthand column here, are all the, for lack of a better word, the numbers that go into the PRA calculation. The initiator frequencies, the component failure rates, and so forth, is listed here.

And thinking about it, a great many of them can be estimated using existing nuclear or non-nuclear information. We point out that a great deal of the zeta that went into the original WASH-1400 study was from non-nuclear sources like this.

And they can be formally combined using Razian statistical methods, which allow you to mix limited sets of operating experience with subjective judgments.

Last but not least, is the list of formal expert elicitation on this. I would point out, these are currently done for large light water reactor PRAs as well.

That's where we get numbers like the frequency of large break LOCAs, is through an elicitation process.

Common cause failures, we have good models, such as the Alpha Factor model that's been used like this. We have very good generic information that's been developed over a number of years.

One of the things that I would point out, what is interesting about the generic common cause failure data is it's stability in the sense that numbers don't change among systems too much, or components too awfully much.

So, it's reasonably robust. Yes, Dr. Bier?

MEMBER BIER: (No response)

MR. STUTZKE: Is there a question from Vicki?

MEMBER BIER: Yes. I had to unmute. Sorry. I just wanted to chime in again on the topic of expert elicitation.

And, I think this is again for background. Is something that needs to be -- that needs to be incorporated in this document.

But, over time there is a lot of evidence that not all experts are equally good at putting what they know into probabilistic terms.

So, at some future time, the Committee or the Agency may want to look into updating the guidance on expert elicitation. But, I don't think that needs to be part of this process today. Thanks.

MR. STUTZKE: Yes, thank you. It reminds me of an interaction I had once with former Commission Apostolakis.

He told me, when he estimated numbers for use in a PRA, he was providing his expert opinion. On the other hand, if I estimated the same number, I was just guessing.

So, you're right. Different experts have different qualifications. And I would agree, we need to revisit our guidance on how to conduct expert elicitation.

Jumping down to the bottom of the list there, human error probabilities, hazard frequencies, external hazard fragilities, none of those require design specific operating experience.

They come from knowledge of the design, the review of the procedures. We have acceptable methods for performing those sorts of analyses.

Any other questions on this slide? Vicki?

MEMBER BIER: (No response)

MR. STUTZKE: Okay. Well, let's go onto slide 44 then. But, I would also emphasize the PRA provides a framework for assessing the uncertainties normally lumped into the parametric uncertainties, the modeling uncertainties, and the completeness uncertainties, like this.

We would certainly expect that people don't just estimate the uncertainty and all the parameters. Do a Monte Carlo propagation up to the final risk metrics and call it a day.

They're actually obliged to understand what factors, which basic events, human error, et cetera, et cetera, are driving the uncertainty in the overall results. So, kind of a decomposition.

But, that's the process that helps you put the uncertainty into perspective. As you can see, things that might be uncertain or questionable, let's say, because of a lack of operating experience.

For example, the turbine trip rate, or an uncomplicated scram rate. Things that we would normally estimate using a lot of statistical data, but we would lack, because the plan hasn't been built.

Those sorts of events don't tend to be risk significant. And therefore, they won't overall have too great an impact on the final risk calculation or the uncertainty in that risk calculation.

Okay. Comments on that?

MEMBER DIMITRIJEVIC: Yes, Monty, I have. Okay. I really appreciate, you know, how you well summarize the usefulness of the PRA. That was really good.

But, the new list of all the PRA applications, the only one with actually level three results were necessary for any of the report, and some -- and most of the design certification which have been submitted now, are not required to have a level three PRA.

So, now when we came to the -- to this slide of uncertainties, you also nicely summarize the positivity. Because one of the main issues with using PRA is associated with uncertainties.

So, we don't even have a good way to address every item that was in the PRA. Which completely addressed modeling uncertainties or completeness uncertainties.

And those uncertainties are still open. Now, when we open these to the level three PRA, would you agree with me the uncertainties multiply like, you know, half the times responded with level one PRA.

And doing the level three, especially when you don't have a location, increases uncertainties associated with the results significantly.

And this is my main objection. Is why do we want to introduce these QHOs when we actually really, you know, the drop are uncertain?

MR. STUTZKE: Well, I would respond, you know, we've done -- the staff has done extensive looking and the state of the art reactor consequence analysis, the SOARCA Project.

Which included a full propagation of uncertainty all the way through the MELCOR and the MAX codes. And the uncertainties were perhaps not as big as one would expect.

All right. The other thing is that the Commission's safety role policy statement, while it was being developed, considered how to decide whether somebody had met the goal.

And after a lot of discussion, they concluded the best way was to compare the mean of the uncertainty distribution to the care chart as well as, you know, then later consider the uncertainties as I've described here on slide 44.

So, from that perspective, you know, the issue was debated a while back. And reasonably result --

(Off record sound interruption)

MEMBER BLEY: Okay. If the members on the public on the line can mute your phones.

(Off record sound interruption)

MEMBER BLEY: Quynh, if you can help us out.

MR. NGUYEN: The members on the public line with a radio or music, could you please turn it off?

MEMBER BLEY: Thank you. Marty, can you hear me now?

MR. STUTZKE: (No response)

MEMBER BLEY: Or did Marty drop off?

MR. STUTZKE: I am here Dennis.

MEMBER BLEY: Oh good. Okay. I want to take you back to your slide 41. You've got me curious, and I started digging around.

I really like this. But, I'm remembering back some time in the last year or so, maybe it was two, because I think we were in person back then in Rockville.

But, we had a meeting with the staff and with representatives of the committee developing this non-light water reactor PRA standard. And we challenged them that there was no guidance here like in other places, on how to structure this search.

And I think we said a structured systematic process. And what date is the version on the standard that you have?

Because I've got the one we reviewed back then, with a 2020 date on it. And I can't find any of this in there.

And the representatives from there said that if you think about it, that they couldn't speak for the committee. So, maybe it's just recently been developed.

But, the bottom line is, I'm glad to see it's going to be here. But, I don't think it was there a year and a half ago or so, whenever we had that meeting.

But, I'm pleased it's here, so.

MR. STUTZKE: Yeah. I don't know about that. The final version of the standard that was issued in February 2021.

MEMBER BLEY: Well, that's really great. And they sent me a bunch of papers that are right in line with this, as to how they've used it in other analyses.

And they put it in here. So, I'm glad to see. I guess it's under initiating events?

MR. STUTZKE: Right.

MEMBER BLEY: We'll have to get their version.

MR. STUTZKE: Yes.

MEMBER BLEY: But, use this -- is this their interpretation?

MR. STUTZKE: Yes. This is a compilation of the various supporting requirements.

MEMBER BLEY: Okay. Derek, we'd like to get that. We're out of date. Thank you.

MR. WIDMAYER: I heard you.

MR. STUTZKE: Well, with that, I'll turn it back over to Bill. I do have some backup slides on PRA and risk metrics if you would like me to discuss any of them.

The origin of the QHOs, the risk surrogates, things like that.

MEMBER BLEY: All right. I think some of us would be interested. But, unless other members really want to see it, I don't think we'll go to that today. I don't think it's terribly relevant.

Bill mentioned that you have some prepared presentations on some of the materials that you guys have talked about today. And I just wondered, is all of that something you could bring and present?

Or is some of this available in some kind of self study modules at the Commission?

MR. RECKLEY: It is. It's publically available. Most of it, I don't know how it could be -- I don't know if it would be effective in self study mode.

MEMBER BLEY: Okay.

MR. RECKLEY: We can -- we can provide it to you, and you can maybe help us assess --

MEMBER BLEY: If you can get that to Derek, I'll take a look. And then we can upload it to the rest of the members and see if anybody wants this in study mode.

MR. RECKLEY: Okay. We will provide that.

MEMBER BLEY: Thank you.

MEMBER KIRCHNER: Dennis, this is Walt. Let me ask you a question.

Based on what Marty's just presented, does this address what you've often stated, starting with a blank sheet of paper, and doing a completed search -- well, not a complete search, but a well-informed search or initiating events, and defining design basis events?

MEMBER BLEY: Yes, it does. Although on the slide, it wasn't quite expressed that way. And I don't know what it says in the standard. But, we better look at that sometime soon.

But, if you remember, we had that meeting on the standard. They were going to come back at some point.

And we talked about this issue. And the representative that was there, I used to work with, sent us about 20 papers dealing with these issues.

And they're the same kinds of things I was putting together in that White Paper we covered. So, maybe it will save me some effort and not going to read it.

But yes, it's supposedly reentered if it's not the same.

MR. RECKLEY: Okay. We have a few more slides to finish up. But, I guess before getting into that, I'll apologize on the single failure discussions.

It's the danger of trying to do by examples. But, I'll also look to see if there's some clarification.

And I know, for example, I was only

involved on the periphery, but the design review guide that the ACRS did look at, it talks about INC, and whether INC is developed under an LNP approach, or the more traditional single failure.

I think both of those avenues are addressed within that design review guide again, that the ACRS has looked at.

So, I'll gather up some examples of that as well. And then maybe even some past examples.

MEMBER BLEY: Was that -- was that the read list of Chapter 70 SRP? Or --

MR. RECKLEY: Yes.

MEMBER BLEY: Okay.

MR. RECKLEY: Yes, that. So, finishing up on the last few slides under design and analysis. We did revise the guidance from the first time that ACRS looked at it.

This is consistent with what we brought to the subcommittee. And for the purposes of design and licensing basis event selection, safety class and SSCs, that other engineering approaches could be used for that.

That was -- this change was a result, or resulted from public stakeholders who wanted to make sure we weren't foreclosing on other generally accepted ways, be they reflected in other guidance documents, IAEA approaches, and so forth.

So, we did change this. I think it really doesn't change much in the way of the overall requirements or approach. Go to slide 46.

We did go through some iterations on some of the specific sections. If you want to go to 47, that's one example.

And again, we brought this before the subcommittee. I'm not sure it was -- felt at that time it warranted a lot of discussion.

But, we did expand and tried to clarify that the -- that an applicant would need to look at the whole range of licensing basis events from AAOs down to very unlikely events.

Go onto 48. This is the DBA. And there was some discussion of this at the subcommittee meeting.

We did add a specific sentence that said for the DBAs they needed to be analyzed from initiation to a safe stable end state. And again, as we've talked several times, assuming only safety related SSCs and safety-related human actions would be credited in that assessment of the DBA.

Any further, or any thought given after

the subcommittee meeting as to whether this kind of scratched that itch that I think was identified in the first iteration?

(No response)

MR. RECKLEY: Okay. Seeing no hands, 49, slide 49. Just kind of wanted to emphasize that we did maintain a fairly traditional safety classification scheme of having safety related, non-safety related but safety significant, which for those more familiar with LNP, those would be non-safety related with special treatment, and non-safety significant SSCs.

And you can draw parallels between that and some of the other approaches like regulatory treatment of non-safety systems. Or the primary, the three prim -- three of the four risk categories in 50.69.

And even to some degree, some similarities, when you start to look at IAEA, specific safety requirements and the introduction of design extension conditions.

So, if there's no questions on safety classification, we can go to slide 50.

MEMBER KIRCHNER: Is there a --

MR. RECKLEY: Go ahead.

MEMBER KIRCHNER: Bill, this is Walt again. Is there any need to make it clear somehow that like if you're invoking the LNP approach, the PRA and all the rest that, I -- you said it, and I can't remember it.

They don't -- the middle category, they call it non-safety related but risk significant? Or -- I can't remember.

MR. RECKLEY: Yeah. Under LNP or in NEI 1804, it's called non-safety related with special treatment.

MEMBER KIRCHNER: Special treatment. Sorry, I misspoke.

MR. RECKLEY: And it really is equivalent to what we're calling non-safety related but safety significant.

In that what we -- what we'll ultimately say is needed for any SSC that's designated as non-safety related but safety significant, is the definition of what is needed in terms of special treatment.

Be it hardware requirements like the environment it needs to withstand, be it relia -- almost certainly a reliability assurance program and measure to carry through in operations.

And maybe even programmatic requirements in terms of inspections, procedures to operate the equipment. Whatever special treatment is needed in order to ensure that that SSC --

MEMBER KIRCHNER: Right.

MR. RECKLEY: Would have the capabilities, the reliabilities, the availabilities that are assumed in the assessments.

So --

MEMBER KIRCHNER: Okay. And so, the -- obviously then, these will make your definition table some place.

MR. RECKLEY: Yes.

MEMBER KIRCHNER: And then we won't have to deal with the many other terms. At least in terms of 53, at least.

And we would -- we would just have these three. We wouldn't have the two by two box. We wouldn't have other -- other terminology then in 53. We would be self-consistent.

MR. RECKLEY: Yes. That's the goal.

MEMBER KIRCHNER: Good. Good.

MR. RECKLEY: Yeah. And there were certain terms that we avoided on purpose, just to not carry forth the confusion for another 50 years.

So, with that, I think we can -- if there's no more questions on that. Analytical margins for operating flexibilities, we've talked about this a fair amount.

And we've really not changed the language very much. Or at all really. This is the provision, this is the section that would define how the analysis needs to be carried through and maintained to support something like the calculation that you could have a smaller emergency planning zone, or you could justify an alternative to the population density criteria in the siting reg guide.

Any other operational flexibilities that we're going to start to get into in Subpart F, this provision is allowing the margins to be traded off.

And then establishing the requirements to make sure that all the assumptions and analysis that went into justify trading off the margins, are maintained over the life of the plan.

So, we can go onto 51, I think. There were really no changes or much of a discussion with external stakeholders or with the ACRS subcommittee on the need to have quality assurance for the design process, and the need to set up interfaces between the design process and things like construction, fairly obvious, operations, and so forth.

So, I don't -- I won't spend much more time on that. Slide 52 goes to non-radiological hazards. And we talked about this with the subcommittee.

We're just -- we are still looking at this and looking at other examples like fuel cycle facilities, to see how we should bring in the non-radiological hazards into Part 53.

We'll acknowledge that it warrants, I don't know, reviewing that topic. And we're currently doing that. And we will come back to the ACRS on our resolution of that.

And with that, I think that's the last slide. Yes.

MEMBER BLEY: Okay. Bill?

MR. STUTZKE: Yes, Dennis?

MEMBER BLEY: I might go back to Marty if he's still with us. Marty, could you pull up your slide 64 and then 65?

MR. STUTZKE: Yes.

MEMBER BLEY: My first question is just personal curiosity on 64. The surrogates here, the QHOs, my memory is that this stuff was put together by Trevor Bott (phonetic) and maybe John Lanoff (phonetic), but both may have been by going to existing PRAs and kind of summarizing this.

Do you remember if that's how that came about?

MR. STUTZKE: Yes. That is.

MEMBER BLEY: And then --

(Simultaneous speaking)

MR. STUTZKE: And when a --

MEMBER BLEY: Go ahead.

MR. STUTZKE: Yeah, that's how it was done. I mean, they actually described it pretty well in the Appendix D of the technology neutral framework, NUREG-1860.

MEMBER BLEY: That's why I remember it. Thank you.

MR. STUTZKE: Yes.

MEMBER BLEY: Go onto 65. There's some real key stuff in this slide. Maybe you can talk us through this one.

And you know, we've -- we thought some about some difficulties. Now Bill's assured us, and we've found they're not too difficult directly of having the QHOs.

But, there are arguments about why it might be better if you use other integral risk measures to test the overall risk for a new plant.

And that you've cited something up here that was also in Appendix in 1860. But, I think in the body of 1860 they went with the QHOs as an integral risk measure.

But in the Appendix they used various other approaches. If you could talk us through that, I think that would be helpful.

MR. STUTZKE: Yeah. It's an historic issue as you said, Dennis. They ended up using the QHOs in the main body.

And there is this extended discussion about the use of complementary accumulative distribution functions, CCDFs, in there. I've cited the main ACRS letter where they debated, you know, the members at that time about the pros and cons of using the method.

The staff deferred action on it. One, because the project was coming to an end. And they wanted to get 1860 published.

The more technical reason is they were worried about anchoring the CCDF to the QHOs. So, the area under the CCDF is the main risk, right. The expected value of risk.

So, you start that way, and the question is then, how do you draw the line? You know, the limit line that the CCDF would represent.

And they never got to that point during the development of NUREG-1860. But rather, it was intended to be deferred until they could pilot 1860 either on the, I think the pebble bed design.

MEMBER BLEY: Yeah. It was the pebble bed. And they backed out.

MR. STUTZKE: Yeah. They backed out and it never got done. It was never picked up.

MEMBER BLEY: So, have you given any thought to relative merits of sticking with the QHOs? Or using something like a CCDF limit curve?

MR. STUTZKE: Yeah. I thought about it occasionally. About how would I come up with the, you know, the shape of the CCDF curve. And make certain it goes through appropriate anchor points and things like that.

But, that's about as far as I've gotten. I'm not ready to say, you know, one way is better than the other.

MEMBER BLEY: Of course Rich Denning and Vinod Mubayi, and Vinod was probably the primary author of that Appendix, wrote a letter to the staff on this very issue.

MR. STUTZKE: Yes.

MEMBER BLEY: I saw a copy of the letter. Is there a response to the letter? Yes, there is. And it's in -- you got it there. That's the end on that one.

MR. STUTZKE: Yes.

MEMBER BLEY: Could you tell us how the staff responded?

MR. STUTZKE: Bill, you want to jump in since I didn't write this response?

MR. RECKLEY: Our response was basically to acknowledge that what Dr. Denning and -- had proposed was a workable approach.

However, we also thought that the LNP, looking at the individual events, and comparing it to the frequency consequence, when combined with looking at the cumulative risk through looking at the QHOs and the other cumulative measure that LNP had provided, that it was also an acceptable approach.

And since for the purpose of writing the regulatory guide, we were being asked to endorse NEI's 1804, that -- that as Marty said, we weren't saying one was necessarily preferable over another. Both could work.

And so that was the response.

MEMBER BLEY: Thank you. We may come back to this some more later. Okay. So, I just wanted to pick up those two.

I don't have anything else. Do any other members have questions or comments you'd like to make before we move to public comments?

MEMBER DIMITRIJEVIC: Well Dennis, I heard two interesting things today. And I was wondering if, are those two informations publically available?

And can we see some of that? Like for example, there was a, I think Bill said there was a -- they've used the QHOs on some of them for Fukushima.

So, that would be something that would be interesting to see. And also, Marty said that the level three is out, which they've done.

I'm not sure I support that right at this moment with the associate uncertainties, show the uncertainties are not so high on level three results.

And I -- if that's a public available book, I would love to see that.

MEMBER BLEY: All right. If the staff could deliver any of that to Derek, that would be helpful. My memory is that back when we reviewed the Fukushima items on the failure to vent for all reactors, that was part of the analysis, was that.

But Bill, I'm not positive of that.

MR. RECKLEY: Yeah.

MEMBER BLEY: So were --

MEMBER RECKLEY: Yes, you did. And it was subsequently published in a NUREG. And so we'll give you that reference.

MEMBER BLEY: Okay. I think that would be helpful. And that would be great for that.

MR. RECKLEY: Yes.

MEMBER BLEY: Okay. Thank you. And Derek, you're on the line. I remember we got the comments from Rich and Vinod.

Did we also get the staff response? Or can you get that for us?

MR. WIDMAYER: I think I got it. But, I'll check and make sure.

MEMBER BLEY: Okay. Thank you.

MR. WIDMAYER: Yeah.

MEMBER BLEY: Anyone else?

(No response)

MEMBER BLEY: At this time I'd like to open the public line. Oh, no, I'm sorry. My day has gone blank.

But, at this time I'm going to invite Cyril Draffin from the USNIC to speak. They requested time with the meeting to make some comments.

And if you are available, please begin.

MR. DRAFFIN: I am. Thank you very much. I am Cyril Draffin, the Senior Fellow for Advanced Nuclear at the U.S. Nuclear Industry Council. And today's remarks augment the comments we provided at the April 22 subcommittee meeting.

First, it may be premature for the ARC to make a definitive comment. The NRC has stressed that Part 53 permitting language will remain open to change until all parts of Part 53 have been provided and stakeholder comments have been received.

Therefore, it may have a negative impact for ACRS to submit a definitive interim letter to support the current Subpart B and C drafts of the rule. Recognize that only a current portion of the Part 53 language is available and the current language is likely to change.

Second, the U.S. Nuclear Industry Council does not agree with the second iteration of Subparts B and C. There are many areas where the preliminary language in Subparts B and C are increasing regulatory burden over Parts 50 and 52.

And the NRC has basically made no major changes to address the industry concerns about those two Subparts.

Also, there is a -- the NRC staff had promised that Subpart F would enable a significant reduction in operational burden as compared to Parts 50 and 52.

And therefore, that justified the increased burden in Subparts B and C to obtain those operational duties, the operational burden.

But, now having seen Subpart F, it's not clear what the benefits are. The preliminary language seems to result in increased burden, doesn't -- still limits flexibility, and doesn't really enhance the safety.

So, we're hoping that the NRC will be receptive to incorporating some of industry's stakeholders' inputs in the coming months.

The only apparent benefit of Part 53 so far, is that there's no need to seek exemptions to large LWR specific requirements.

Then a few points that we've covered before, which I think are still relevant, particularly for people that weren't on the subcommittee meeting.

For the adequate protection standard, we disagree with the second revision to the strategic objectives that drop the formal reference to reasonable assurance of adequate protection standard.

We think that it, adequate protection to public health and safety is important. And changing the objectives primarily to justify the preliminary language seems questionable.

For the tiers, we still think the tiered category one and two are confusing, with opportunities for unintended consequences. The second rendition of the SOARCA objectives drops the language in the Atomic Energy Act, and so the same 51 and 52 seem less relevant.

And we might consider a simple tier unless the operational language shows real benefits, and particularly for all the criteria discussed earlier that have to be met.

We continue to believe Part 53 should be technology inclusive to allow both risk based and deterministic methods. And that it should not be limited to just applications using the PRA tool, although it's a very valuable tool

And with this second iteration, it's still too restrictive in requiring a PRA. As discussed earlier, we think PRA should be applicable for a range of licensing path and technologies.

And that risk insights are what are

important for design, not the specific numerical results. We don't believe that PRA should be elevated to a compliance tool as part of the application, especially for a construction permit.

It's not clear that any approach used by Oklo or NuScale would comport with a prescriptive use of the PRA as a compliance tool.

And if it is included, as it is not, in Part 53 as a requirement, then exemptions will be required for some of the technologies, which seems a little inconsistent with the original goals and objectives of Part 53.

Now, the timing for a phased or simplified approach has merit. And I think that there's some flexibility on how that's done. It merits further discussion.

For ALARA, many stakeholders, as mentioned this morning, believe that ALARA is an important concept and certainly a good practice that we expect to continue.

But, we do not believe ALARA should be included in Part 53 formal regulation in part because of the subjectivity and complexity of ALARA in the design phase. New operation should be like protection of plant workers and should not be included in the safety criteria.

And for defense in depth, it's important as a design philosophy and supporting an adequate safety case. But, the defense in depth details should be in guidance and not added to the regulations.

We believe that Part 53 can have predictability as well as flexibility. We think it really can have specific performance criteria that must be demonstrated, and flexibility to allow them to be made and not just relying upon LNP as the process.

And finally, we do support the consensus codes and standards, which are being adopted by NRC. So, those are some comments for you to consider as you draft your interim letter.

Thank you for the opportunity.

MEMBER BLEY: Mr. Draffin, thank you very much. You will be on the transcript. And you will have access to that.

But, if you prefer to also send your comments in writing, both they -- I shouldn't say it won't be garbled, but once in a while, transcripts don't read exactly like you thought you said them.

So, if you wanted to do it in writing too, that's fine. Just give them to Derek.

MR. DRAFFIN: Well, thank you.

MEMBER BLEY: So, Thomas or Makeeka, can we get the public line open?

MR. DASHIELL: The public line is open for comments.

MEMBER BLEY: Thank you very much. If there's anyone who would like to make a comment, please identify yourself and make your comments.

(No response)

MEMBER BLEY: All right. Okay, I think we can close the public line at this point.

For the members, we had a very long session last week of deliberations. And as a result of that, I really thank everyone for all the ideas and written suggestions and the discussion.

But, it helped a lot. As I began to organize my notes from it, that session, the pieces started to come together.

And I think they're -- they're still pretty much holding out that there might be some areas we'll have to dig into. As I drafted a letter, I tried to include areas where I had a sense we had agreement.

And for other areas, rather than reconcile, I tried to integrate that, or put together the areas of concern. I couldn't address every issue that everyone raised. We don't need that in this letter.

Added comments may be necessary or are certainly welcomed. But that's kind of irrelevant. You can write them anyway. But any area where one or two people feel very strongly about, maybe it's important.

At this time, I think we'll call this meeting to an end. But, we'll move into a letter writing session if that's okay, Matt.

But we will go off the record. But, I'll let you do that since this is a full committee meeting.

CHAIR SUNSERI: Thank you, Dennis. So, could we take like a ten minute break though before we go into reading the letter?

Would that be okay to everyone?

MEMBER BLEY: Yeah. I was going to suggest that. And I think we'll try to finish in 15 or 20 minutes after.

I don't know, if we really get into discussion, it could take a long time. But, I'd like everybody to hear where it stands now and be able to read it later.

I think Derek will verify this. I think

he's got it up on the website. But, I'm not quite sure.

CHAIR SUNSERI: Okay. Very good. Then we'll take a 15 minute break here. We'll recess until 6:15. Is it 6:15, is that right? Yeah.

Oh, it's 6:00. Okay. All right. We'll recess until 6:15. And then we'll pick it up and read through the letter and finish today at the conclusion of that activity.

All right. So, we are recessed.

(Whereupon, the above-entitled matter went off the record at 6:00 p.m.)

Advisory Committee on Reactor Safeguards (ACRS)

10 CFR Part 53
"Licensing and Regulation of
Advanced Nuclear Reactors"

May 5, 2021

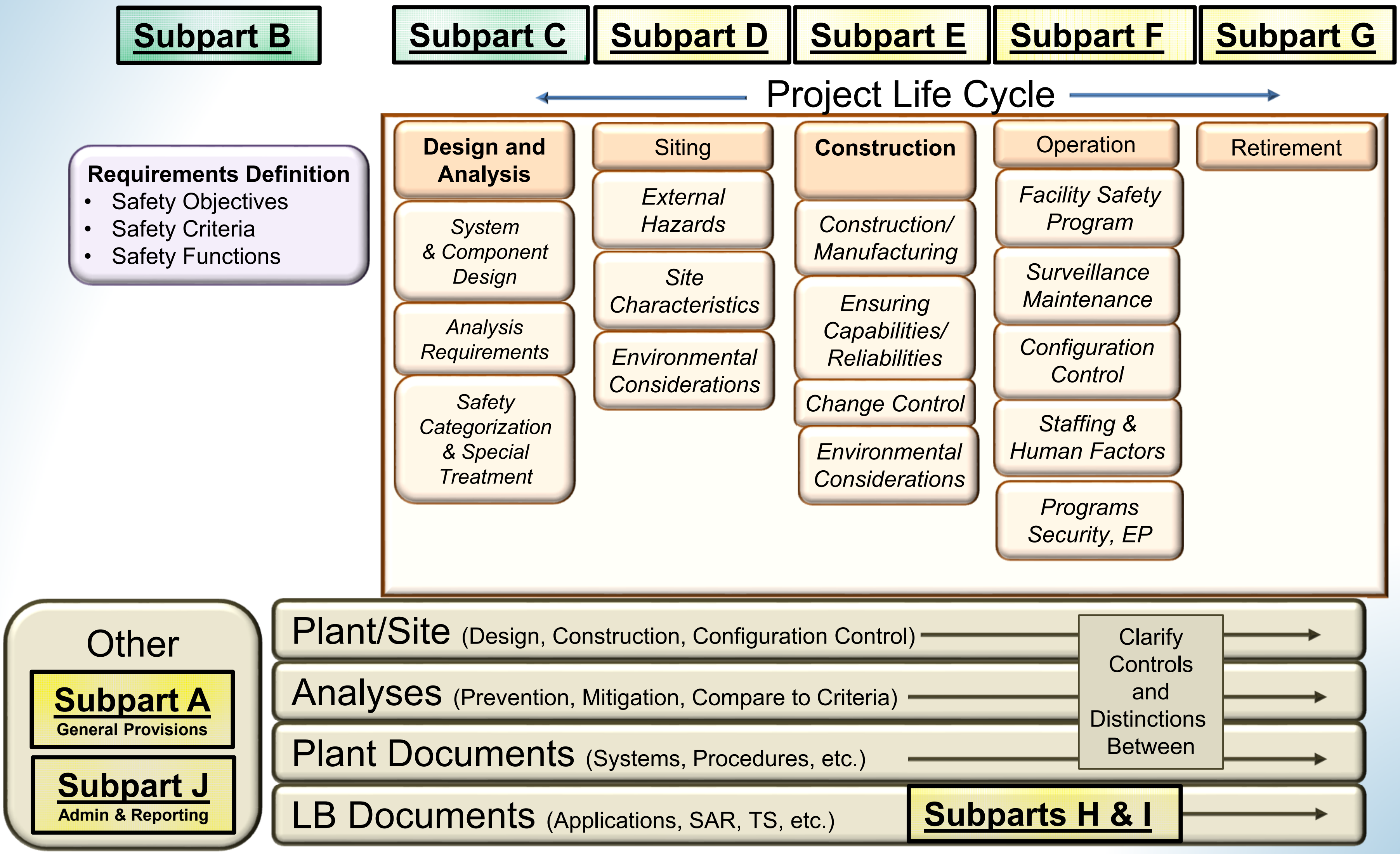
Agenda

- Opening Remarks
- Overall Structure (Framework)
- Subpart B – Technology-Inclusive Safety Requirements
- Subpart C – Design and Analysis Requirements
- Discussion

Background

- Nuclear Energy Innovation and Modernization Act (NEIMA; Public Law 115-439) signed into law in January 2019 requires the NRC to complete a rulemaking to establish a technology-inclusive, regulatory framework for optional use for commercial advanced nuclear reactors no later than December 2027
  - (1) ADVANCED NUCLEAR REACTOR—The term “advanced nuclear reactor” means a nuclear fission or fusion reactor, including a prototype plant… with significant improvements compared to commercial nuclear reactors under construction as of the date of enactment of this Act, …

NRC Staff Plan to Develop Part 53



Part 53 Contents

(A)	General Provisions *(including definitions)*
(B)	**Safety Criteria *(two tiers/categories, as low as reasonably achievable (ALARA), defense in depth (DiD)***
(C)	**Design and Analysis *(design criteria, role of probabilistic risk assessment (PRA))***
(D)	Siting *(external hazards, population)*
(E)	Construction and Manufacturing *(factory fueling)*
(F)	Operations *(structures, systems and components (SSCs), staffing, programs)*
(G)	Decommissioning
(H)	Licensing *(siting, design, licenses)*
(I)	Maintaining Licensing Basis
(J)	Administrative and Reporting

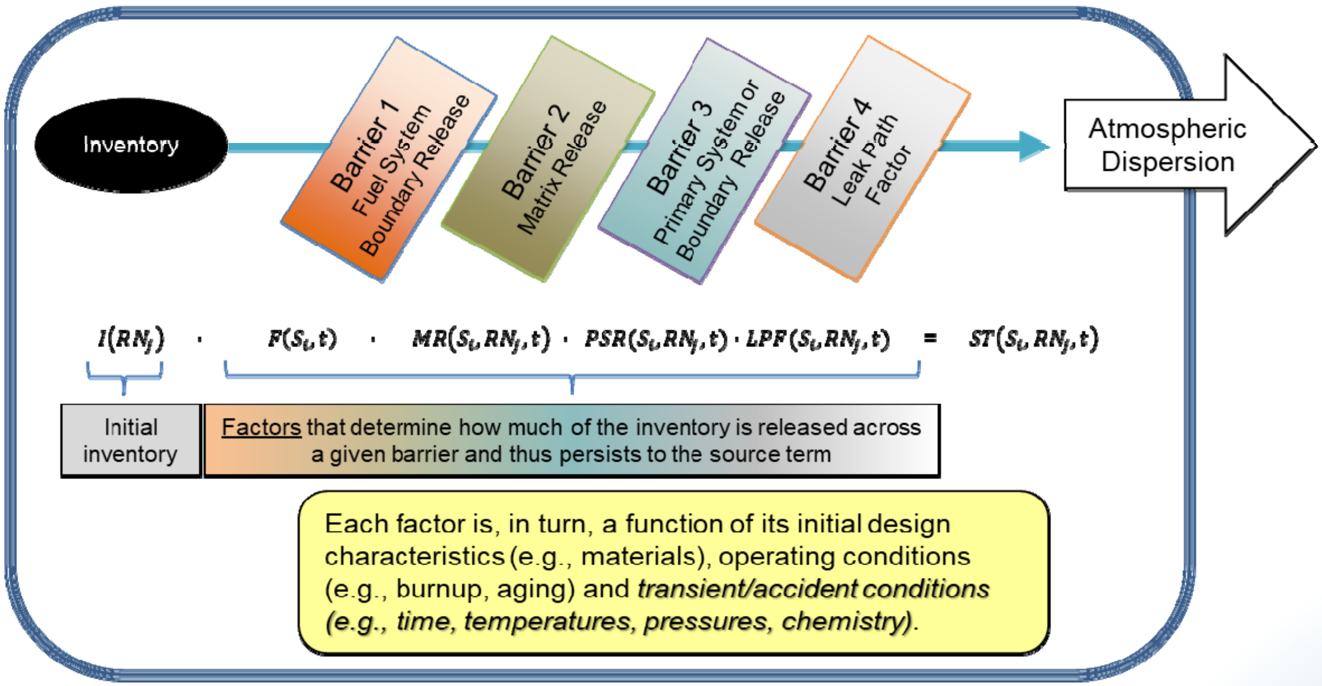
Subpart B
Technology-Inclusive Safety Requirements
Preliminary Language

Rulemaking Plan (SECY-20-0032)

The staff plans to build upon ongoing activities such as those described in SECY-19-0117, "Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," dated December 2, 2019 (ADAMS Accession No. ML18311A264), **to develop the associated performance criteria**. … The methodology described in SECY-19-0117, … includes identifying the potential benefits provided by design features and programmatic controls in terms of the margins between estimated doses and the reference values in NRC regulations and the margins between estimated health effects and the NRC's safety goals. SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors," dated September 28, 2018 (ADAMS Accession No. ML18115A157), and SECY-18-0103, "Proposed Rule: Emergency Preparedness for Small Modular Reactors and Other New Technologies (RIN 3150-AJ68; NRC-2015-0225," dated October 12, 2018 (ADAMS Accession No. ML18134A076), provide examples of how those margins are used within performance criteria for potential operational flexibilities.

First Principles

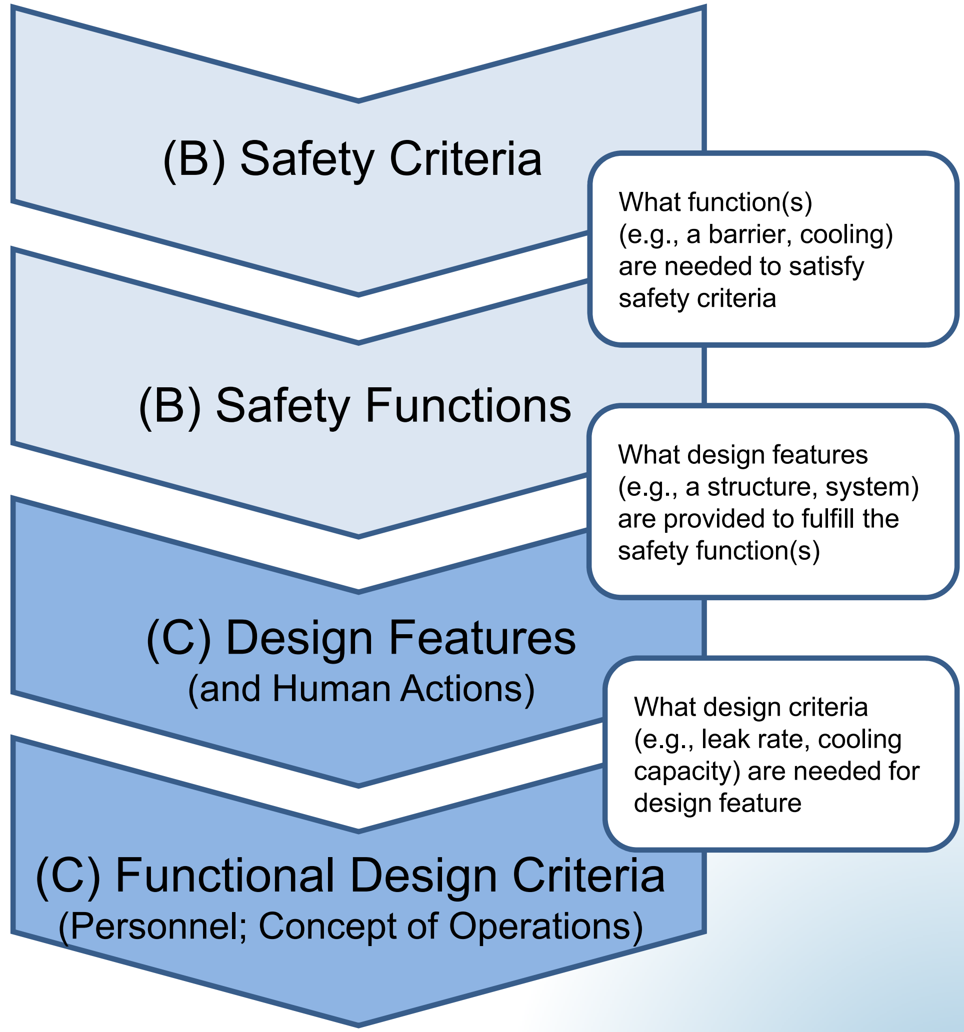
Recent NRC activities related to advanced reactors (e.g., functional containment performance criteria, possible changes to emergency planning & security, and DG-1353) recognize the limitations of existing LWR-related guidance, which requires a return to first principles such as fundamental safety functions supporting the retention of radionuclides



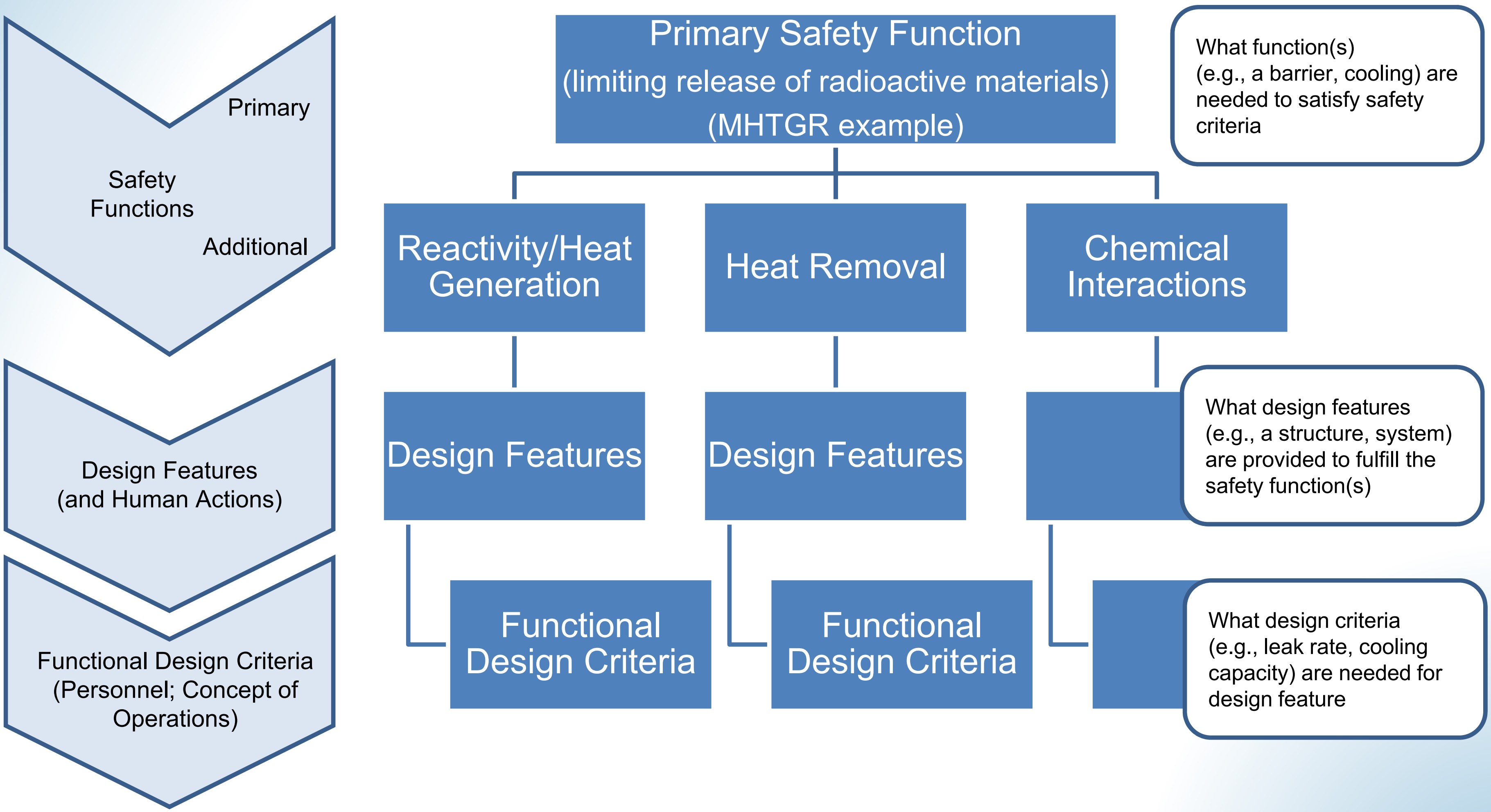
See: **SECY-18-0096**, "Functional Containment Performance Criteria for Non-Light-Water-Reactors," INL/EXT-20-58717, "Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities," and **SECY-19-0117**, "Technology-Inclusive, Risk-Informed, and Performance-Based Methodology.."

Subpart B – Safety Criteria

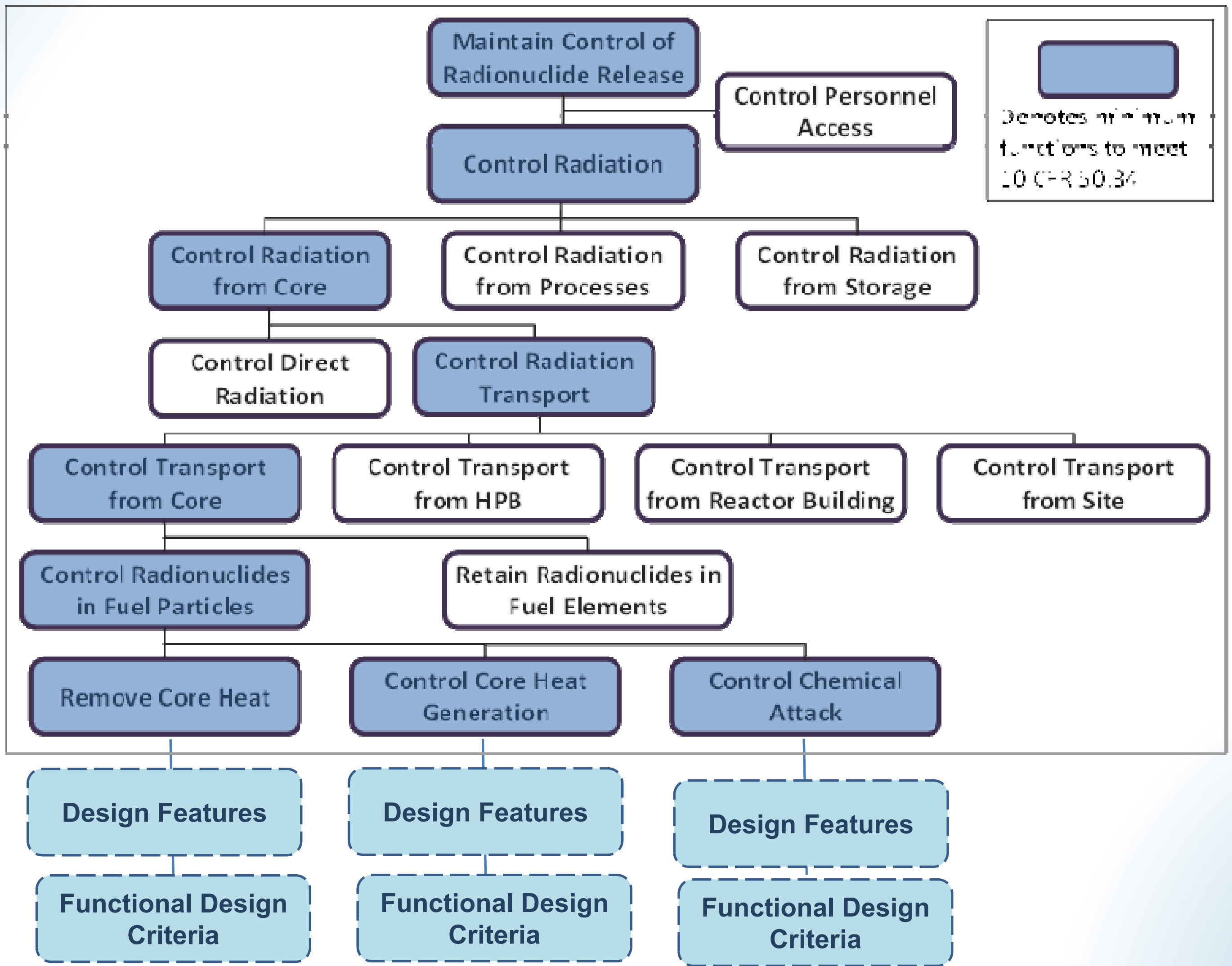
- Safety Objectives
- First Tier Safety Criteria
  - Immediate threat to public health and safety
- Second Tier Safety Criteria
  - Appropriate to address potential risks to public health and safety
- Safety Functions
- Licensing Basis Events (LBEs)
- Defense in Depth
- Protection of Plant Workers



Technology-Inclusive Methodology



Modular High-Temperature Gas-Cooled Reactor (MHTGR) Example (Safety Functions)



Addressing Functions & Design Criteria

(B) Safety Functions

(C) Design Features

(C) Functional Design Criteria

10 CFR 50, Appendix A
General Design Criteria

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Addressing Functions & Design Criteria

(B) Safety Functions

(C) Design Features

(C) Functional Design Criteria

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	Monitoring Radioactivity Releases	64

Part 50 and Part 53
Comparing Licensing Frameworks

- Safety criteria
  - Same safety criteria in Parts 50 and 53
  - Quantitative health objectives (QHOs) used in guidance under Part 50
- Design and Analyses
  - Design Basis Accidents (DBAs)
    - Part 50: Assessed using prescriptive, highly conservative analyses
      - Including single failure criterion (SFC)
    - Part 53: Assessed methodically considering event frequencies and assuming only safety-related SSCs are available
  - Beyond Design Basis Events (BDBEs)
    - Part 50: Identified & assessed by largely ad-hoc, prescriptive approach with uncertainties addressed through conservatisms
    - Part 53: Derived methodically using event frequencies with explicit consideration for uncertainties
      - Including combinations of various equipment failures
- Special Treatment for Non-Safety-Related but Risk-Significant SSCs
  - Part 50: Ad-hoc (e.g., § 50.69 programs, Reliability Assurance Programs (RAP))
  - Part 53: Systematic approach to control frequencies and consequences of the LBEs in relation to safety criteria

Second Iteration – Objectives

§ 53.200 Safety Objectives.

Each advanced nuclear plant must be designed, constructed, operated, and decommissioned to limit the possibility of an immediate threat to the public health and safety. In addition, each advanced nuclear plant must take such additional measures as may be appropriate when considering potential risks to public health and safety. These safety objectives shall be carried out by meeting the safety criteria identified in this subpart.

- Discussion
  - Generally aligns with requirements for content of technical specifications and regulatory treatment of non-safety systems
  - Addresses concerns related to tying tiers to authorities provided in the Atomic Energy Act (e.g., adequate protection and minimize danger to life or property)

Second Iteration – First Tier

§ 53.210 First Tier Safety Criteria.

(a) Public dose does not exceed Part 20 limit (0.1 rem) from normal plant operation

(b) Provide design features and programmatic controls such that events with frequencies greater than once per 10,000 years meet the following

(1) 2-hour dose below 25 rem at EAB

(2) Duration dose below 25 rem at LPZ boundary

- Discussion
  - Maintains technical criteria from first iteration
  - Generally aligns with requirements for content of technical specifications and regulatory treatment of non-safety systems
  - Deleted paragraph (c) since the first tier criteria are no longer tied to adequate protection standard
  - Added existing footnote on 25 roentgen equivalent man (rem) as reference value
  - General note that staff assessing terminology (tiers)

Additional Discussion – First Tier

- Possible Applications of First Tier Safety Criteria
  - Minimally acceptable level of safety
  - Met by satisfying the safety functions needed for dose < 25 rem
  - Provides basis for safety classification of safety-related SSCs
  - Demonstration of meeting the first tier safety criteria supported by analyses of DBA
  - Provides basis for identifying SSCs needing protection against external events up to the design basis external hazard levels
  - Provides basis for identifying appropriate content of technical specifications (TS)
    - Reserved for the most significant safety requirements
    - Necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety
  - May provide basis for staffing and operator licensing decisions
  - Greatest level of detail for information in licensing documents

Second Iteration – Second Tier

- Second Tier Safety Criteria

FIRST ITERATION/SECOND ITERATION
§ 53.220 Second Tier Safety Criteria.

(a) *Normal operations*. Design features and programmatic controls must be provided for each advanced nuclear plant to ensure the estimated total effective dose equivalent to individual members of the public from effluents resulting from normal plant operation are as low as is reasonably achievable taking into account the state of technology, the economics of improvements in relation to the state of technology, operating experience, and the benefits to the public health and safety. Design features and programmatic controls must be established such that [to be reworded for consistency with 10 CFR part 20 and 40 CFR part 190].

(b) *Unplanned events*. Design features and programmatic controls must be provided to:
(1) Ensure plant SSCs, personnel, and programs provide the necessary capabilities and maintain the necessary reliability to address licensing basis events in accordance with § 53.240 and provide measures for defense-in-depth in accordance with § 53.250; and
(2) Maintain overall cumulative plant risk from licensing basis events such that the risk to an average individual within the vicinity of the plant receiving a radiation dose with the potential for immediate health effects remains below five in 10 million years, and the risk to such an individual receiving a radiation dose with the potential to cause latent health effects remains below two in one million years.

Feedback – 2nd Tier, ALARA

- ALARA
  - Proposal by some stakeholders to eliminate all ALARA requirements under Part 53.
- NRC Iteration: Maintained requirements for normal operations and occupational exposures to be ALARA

Note that concerns related to ALARA and NRC reviews of design-related applications are also being addressed through the Advanced Reactor Content of Application Project with current drafts of Chapter 9 released to support stakeholder interactions:

"... in lieu of providing detailed system descriptions and analysis of estimated effluent releases as required by 10 CFR 50.34, 50.34a, 52.47, and 52.79, an application may demonstrate compliance with the applicable regulations by describing a radiation protection program and an effluent release monitoring program that will ensure that effluent release limits will be met during normal operations for the life of the plant. Information related to physical systems can be limited to general descriptions of layout and technologies used to limit the release of the various inventories of radioactive materials within the plant."

Feedback – 2nd Tier, QHOs

- QHOs
  - Proposal by some stakeholders to maintain QHOs as policy but exclude from rule
    - Some concern over use of QHOs related to inclusion of requirement to perform PRA
  - Proposal by some stakeholders to use a metric other than QHOs as second tier
    - Range of stakeholder views, from use of QHOs to use of cost-benefit assessment for second tier, which in NRC practice includes assessment against QHOs
- NRC Iteration: Maintained QHOs within the second tier safety criteria
  - The QHOs are a well-established measure used in NRC risk-informed decision making and are a logical performance metric to support the risk management approaches to operations that will be reflected in Subpart F, "Operations."
  - Note that using less defined criteria for the second tier would decrease the predictability of the regulations in terms of the desired graded approach (e.g., differentiation between SSCs that are safety related and non-safety related with special treatment)

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Additional Discussion – Second Tier

- Possible Applications of Second Tier Safety Criteria
  - With first tier, ensures appropriate level of safety for long-term, risk-informed operations
  - Met by satisfying the safety functions for meeting QHOs
  - Demonstration of meeting the second tier safety criteria supported by systematic analyses (i.e., PRA)
  - Provides basis for identifying additional risk-informed requirements
  - Provides basis for identifying appropriate special treatment for non-safety related SSCs (e.g., functional design requirements & reliability)
  - Provides basis for enabling risk management approach to operations
  - May provide basis for staffing and operator licensing decisions
  - Enables appropriate level of detail in licensing basis documentation based on a risk-informed, function-oriented and performance-based approach

Second Iteration – Safety Functions

§ 53.230 Safety Functions

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

(b) Additional safety functions supporting the retention of radioactive materials during routine operation and licensing basis events—such as controlling [reactivity], heat generation, heat removal, and chemical interactions--must be defined.

(c) The primary and additional safety functions are required to meet the first and second tier safety criteria and are fulfilled by the design features and programmatic controls specified throughout this part.

- Discussion (Safety Functions)
  - Maintains mention of fundamental safety functions as examples to maintain technology-inclusive framework (with potential use for multiple inventories of radionuclides within plants and possibly technologies such as fusion energy systems)
  - Reinforces general hierarchy of safety criteria, safety function, design feature, and functional design criteria.

Second Iteration – LBEs

§ 53.240 Licensing Basis Events

Licensing basis events must be identified for each advanced nuclear plant and analyzed in accordance with § 53.450 to support assessments of the safety requirements in this subpart B. The licensing basis events must address combinations of malfunctions of plant SSCs, human errors, and the effects of external hazards **ranging from anticipated operational occurrences to very unlikely event sequences with estimated frequencies well below the frequency of events expected to occur in the life of the advanced nuclear plant.** The evaluation of licensing basis events must be used to confirm the adequacy of design features and programmatic controls needed to satisfy first and second tier safety criteria of this subpart and to establish related functional requirements for plant SSCs, personnel, and programs.

- Discussion (LBEs)
  - Changes to clarify the range of scenarios to be addressed by LBEs

Licensing Basis Events – Light-Water Reactor (LWR) Summary

a. Consider events that can result in the basic parameter changes listed below and identify potential limiting events:
- Increase of core reactivity,
- Changes of reactor coolant flow,
- Changes of reactor coolant pressure,
- Changes of reactor coolant temperature,
- Changes of reactor coolant inventory,
- Changes in energy supplies to the plant,
- Changes in coolant supplies to the plant,
- Changes in the nuclear safety-related equipment status,
- Changes in core power distribution,
- Changes in radioactive releases, or
- Changes of any other variable that has a limiting value.

EVENT FREQUENCY RANGE (per reactor-year)	PLANT CONDITIONS CATEGORIES	OTHER CATEGORIZATION SCHEMES					
			NRC		ANS		
		10 CFR	RG 1.48 ASME Code*	RG 1.70 Rev. 2	51.1 (N18.2)	52.1 (N212)	53.1 (N213)
Planned Operations	PC-1	Normal	Normal	Normal	Condition I	Normal PPC	Plant Condition A
10^{-1}	PC-2	Anticipated Operational Occurrences	Upset	Moderate Frequency	Condition II	Frequent PPC	Plant Condition B
10^{-2}	PC-3			Infrequent Incidents	Condition III	Infrequent PPC	Plant Condition C
10^{-3} / 10^{-4}	PC-4	Accidents	Emergency	Limiting Faults	Condition IV	Limiting PPC	Plant Condition D
10^{-5} / 10^{-6}	PC-5		Faulted				
	Not sidered						

n 1977 version of the ASME Code.

Fig. B-1
Event Categorization

Table 3-1

Offsite Radiological Dose Criteria for Plant Conditions

Best-Estimate Frequency of Occurrence (F) Per Reactor Year	Plant Condition (PC)	Offsite Radiological Dose Criterion
Normal Operations	PC-1	10 CFR 50, App. I(a) [18]
$F \geq 10^{-1}$	PC-2	10 CFR 50, App. I(a) [18]
$10^{-1} > F \geq 10^{-2}$	PC-3	10% 10 CFR 100(b) [2]
$10^{-2} > F \geq 10^{-4}$	PC-4	25% 10 CFR 100(b) [2]
$10^{-4} > F \geq 10^{-6}$	PC-5	100% 10 CFR 100(b) [2]

ANSI/ANS-51.1-1983; nuclear safety criteria for the design of stationary pressurized water reactor plants (withdrawn 1989)

Licensing Modernization Project (LMP): Event Selection & Analysis

- Introduction of an actual frequency-consequence curve as part of the regulatory process (vs. general relationship of decreased consequences expected for more frequent events)

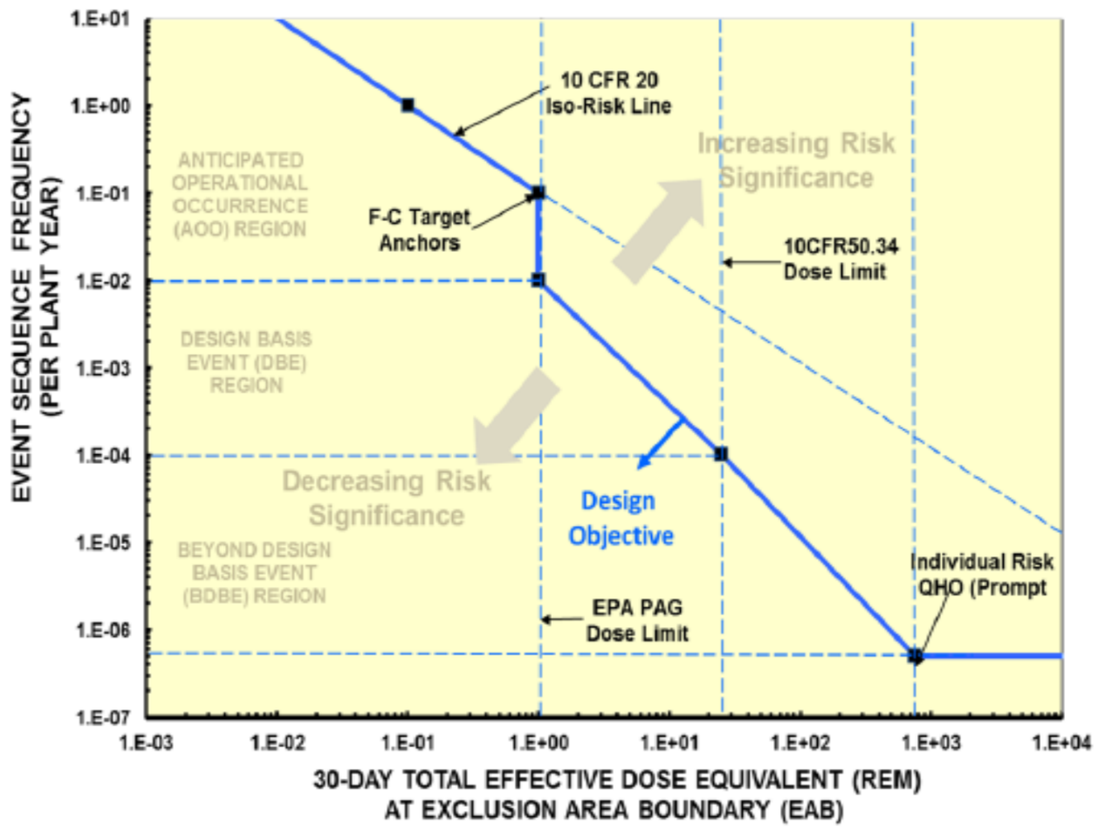


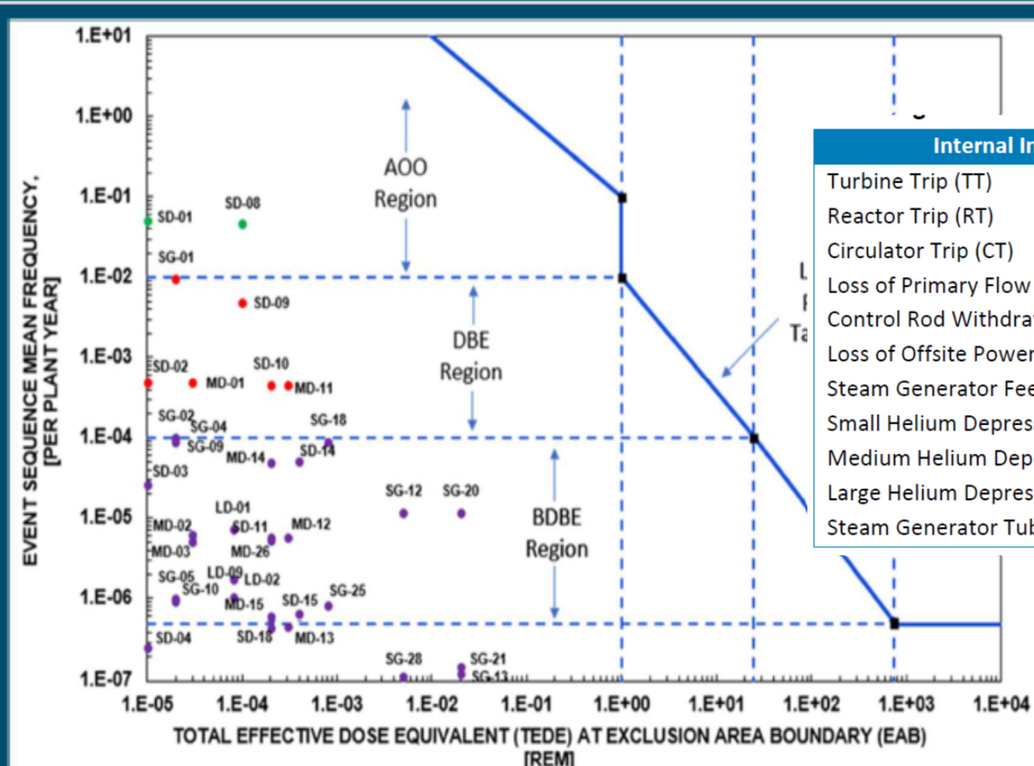
Figure 3-1. Frequency-Consequence Target

Tabletop Exercise (MHTGR; Xe-100)

XE-100 LMP DEMONSTRATION HIGHLIGHTS

NEI

- Example of LMP application at early state of design.
- Limited scope high level PRA developed during preconceptual design to guide conceptual design.
- Completed preliminary selection of LBEs and RSFs with examples identified for SR SSCs.



Internal Initiating Events
Turbine Trip (TT)
Reactor Trip (RT)
Circulator Trip (CT)
Loss of Primary Flow (LF)
Control Rod Withdrawal (CR)
Loss of Offsite Power (LO)
Steam Generator Feedwater Pump Trip (FW)
Small Helium Depressurization (SD)
Medium Helium Depressurization (MD)
Large Helium Depressurization (LD)
Steam Generator Tube Rupture (SG)

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Report: ADAMS Accession No. ML18228A779

LMP: Event Selection & Analysis

Anticipated Operational Occurrences (AOOs)

[Part 53 – AOOs]

Anticipated **event sequences** expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. **Event sequences with mean frequencies of 1×10^{-2}/plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification.**

DBEs

[Part 53 – Unlikely events]

Infrequent **event sequences** that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than AOOs**. Event sequences with mean frequencies of 1×10^{-4}/plant-year to 1×10^{-2}/plant-year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification.**

BDBEs

[Part 53 – Very unlikely events]

Rare **event sequences** that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than a DBE. **Event sequences with mean frequencies of 5×10^{-7}/plant-year to 1×10^{-4}/plant-year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification.**

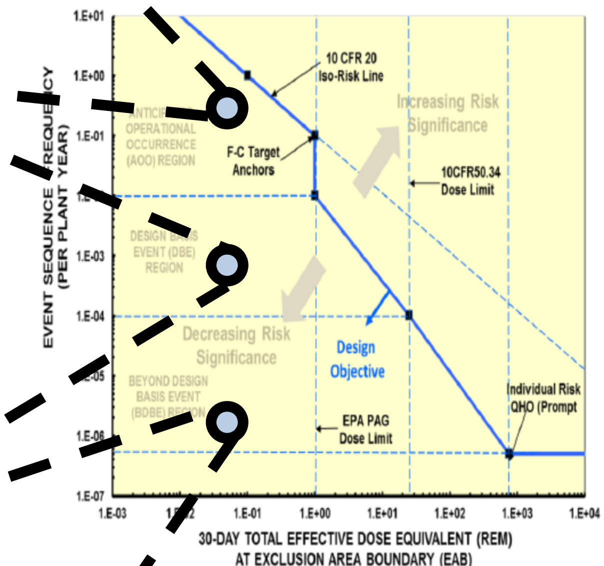


Figure 3-1. Frequency-Consequence Target

LMP: Required Safety Functions

Required Safety Function (RSF): A PRA Safety Function that is required to be fulfilled to maintain the consequence of one or more DBEs or the frequency of one or more high-consequence BDBEs inside the F-C Target

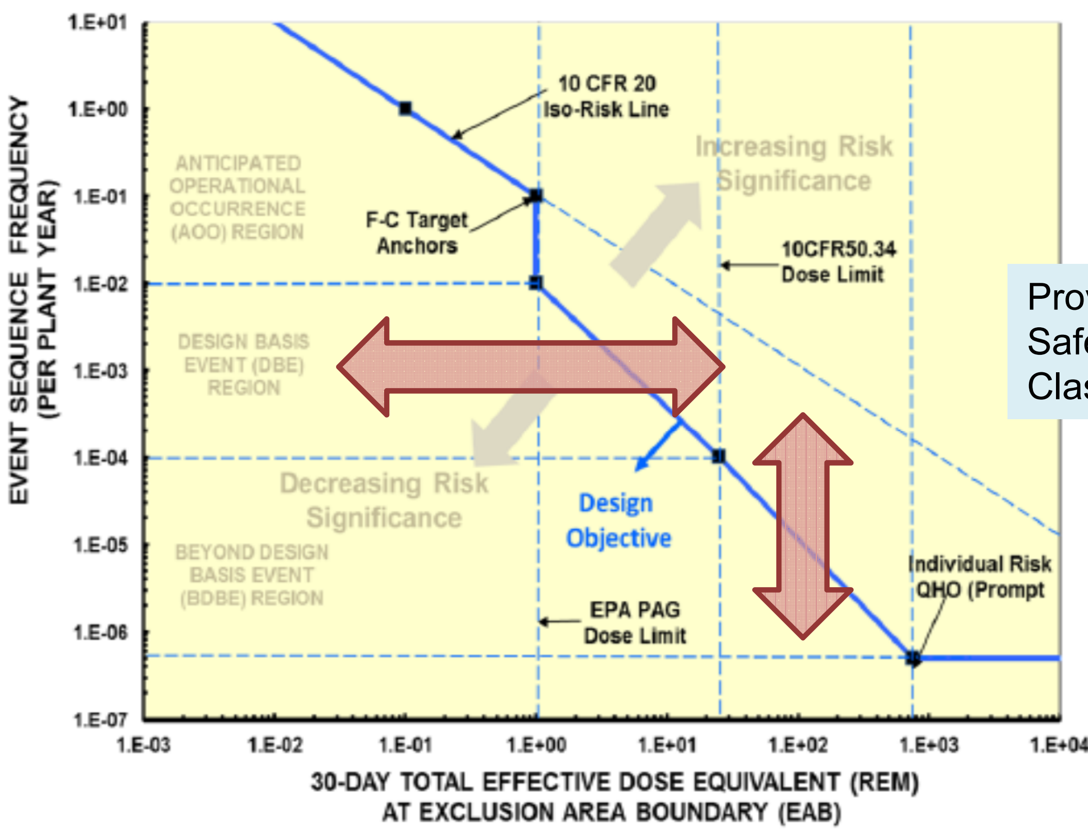


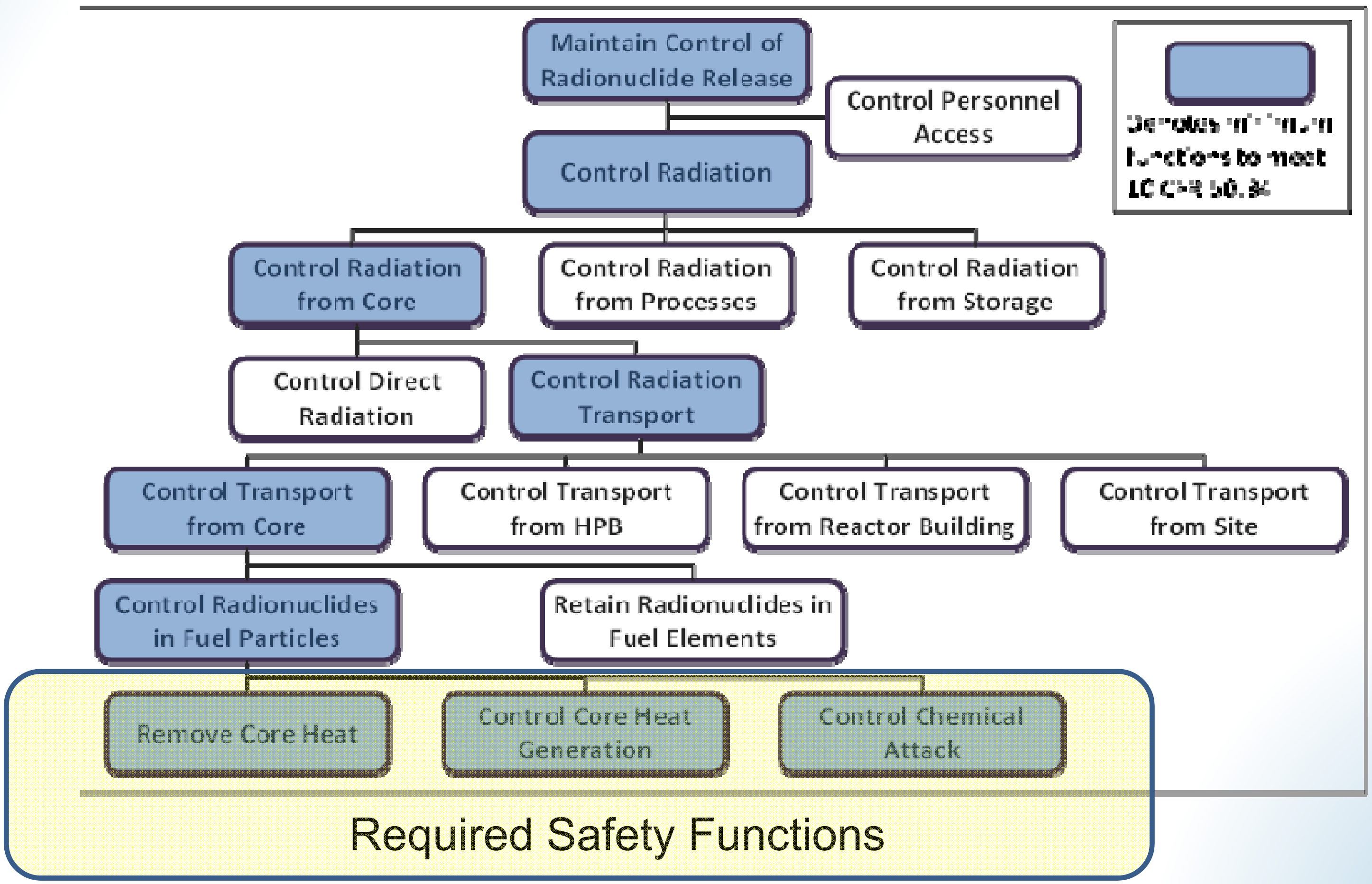
Figure 3-1. Frequency-Consequence Target

Provides connection to Safety-Related Classification

Note – in Part 53, RSFs would translate to those functions needed to address first tier safety criteria

RSF Example

- MHTGR RSFs



Design Basis Accidents

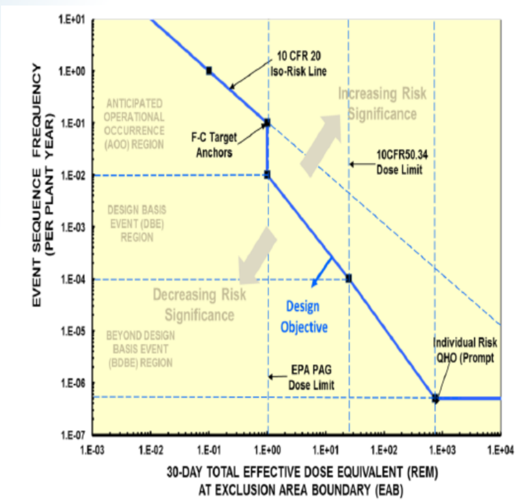


Figure 3-1. Frequency-Consequence Target

DBAs

[Part 53 – DBAs]

Postulated event sequences that are used to set design criteria and performance objectives for the design of Safety Related SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of Safety-Related SSCs needed to mitigate and prevent event sequences, respectively. **DBAs are derived from the DBEs by prescriptively assuming that only Safety Related SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits.**

Second Iteration – DiD

§ 53.250 Defense in Depth

Measures must be taken for each advanced nuclear plant to ensure appropriate defense in depth is provided to compensate for uncertainties such that there is high confidence that the safety criteria in this subpart are met over the life of the plant. The uncertainties to be considered include those related to the state of knowledge and modeling capabilities, the ability of barriers to limit the release of radioactive materials from the facility during routine operation and for licensing basis events, and those related to the reliability and performance of plant SSCs, personnel, and programmatic controls. No single **engineered** design feature, human action, or programmatic control, no matter how robust, should be exclusively relied upon to meet the safety criteria of § 53.220(b) or the safety functions defined in accordance with § 53.230.

- Discussion (DiD)
  - Maintains defense in depth within Subpart B because of historical and continued importance of its role in addressing risk
  - Parts 50/52 do not include a similar section because the defense-in-depth philosophy is incorporated into prescriptive technical requirements for light-water reactors
  - Possibility that this section could be addressed within Subpart C can be considered as part of the later review of the technical requirements
  - Reflects possible crediting of inherent characteristics within the design and analysis for advanced reactors and the reduced uncertainties associated with such characteristics

Second Iteration – Protection of Plant Workers

§ 53.260 Protection of Plant Workers

(a) Design features and programmatic controls must exist for each advanced nuclear plant to ensure that radiological dose to plant workers does not exceed the occupational dose limits provided in subpart C to 10 CFR part 20.

(b) **As required by Subpart B to 10 CFR part 20, design features and programmatic controls** must, to the extent practical, be based upon sound radiation protection principles to achieve occupational doses that are as low as is reasonably achievable.

- Discussion (Protection of Plant Workers)
  - Maintains the protection of plant workers within Subpart B to capture occupational exposures within the high-level safety requirements
  - Changed to refer to part 20, as suggested by stakeholders

Note that ALARA is not only a long-standing requirement by Atomic Energy Commission/NRC (including maintaining in Part 20 rulemaking) but also is addressed in U.S. Environmental Protection Agency Federal Guidance for Radiation Protection

Subpart C
Design and Analysis
Preliminary Language

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Subpart C – Design and Analysis

- Design Features
- Functional Design Criteria for First Tier Safety Criteria
  - Comparable to Principal Design Criteria for Safety-Related SSCs
- Functional Design Criteria for Second Tier Safety Criteria
  - Provides Design Criteria for Safety Significant Non-Safety-Related SSCs
- Functional Design Criteria for Protection of Plant Workers
- Design Requirements
- Analysis Requirements
  - Role of PRA
- Safety Categorization and Special Treatment
- Application of Analytical Safety Margins to Operational Flexibilities
- Design Control Quality Assurance
- Design and Analyses Interfaces

Design-Related Discussions
SFC vs Reliability Criterion

Part 53	PRA Required; Reliability Assurance through TS/RAP Subpart F
RG 1.233 (Licensing Modernization) (SECY-19-0117)	The staff finds that the NEI 18-04 methodology, including assessments of event sequences and DiD, obviates the need to use the single-failure criterion (SFC) as it is applied to the deterministic evaluations of AOOs and DBAs for LWRs.
SRM-SECY-19-0036 (Application of the Single Failure Criterion to NuScale IAB Valves)	The staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria such as the SFC is unnecessary to provide for reasonable assurance of adequate protection of public health and safety.
SECY-03-0047 (Policy Issues Related to Licensing Non-Light-Water Reactor (NLWR) Designs) SRM dated 6/26/2003	The SFC would be replaced with a reliability criterion and the event scenarios identified in the PRA would be examined against this criterion. Note that Issue 4 in SECY-03-0047 also described probabilistic event selection and safety classification

The SFC

- The SFC has the direct objective of promoting reliability through the enforced provision of redundancy in those systems which must perform a safety-related function
- In SECY 77-439 (ML060260236), the staff critiqued the SFC at the request of the Commission:
  - The SFC has served well in its use as a licensing review tool to assure reliable systems as one element of the defense in depth approach to reactor safety.
  - The SFC is just one of several tools applied in systems design and analysis to promote reliability of the systems which are needed in a nuclear power plant for safe shutdown and cooling, and for mitigation of the consequences of postulated accidents. It is not sufficient by itself.
  - The SFC was developed without the benefit of numerical assessments on the probabilities of component or system failure.
  - The Reactor Safety Study (WASH-1400, the first nuclear plant PRA) also pointed out that factors such as systems interactions, multiple human errors, and maintenance and testing requirements also have an influence on reliability. Such factors fall outside the scope of the SFC, and supplementary methods must be utilized In their study.
  - It is expected that probabilistic methods of the type used in the Reactor Safety Study will gradually come into increasing use and supplement the SFC.
- See also the discussions in SECY-03-0047, SECY-05-0138, SECY-19-0036, SECY-19-0117 and related SRMs

Codes and Standards

§ 53.440 Design Requirements.
(a) The design features required to meet the first and second tier safety criteria defined in §§ 53.210 and 53.220 shall be designed using generally accepted consensus codes and standards wherever applicable.

Preliminary Definition (Subpart A): Consensus code or standard means any technical standard (1) developed or adopted by a voluntary consensus standard body under procedures that assure that persons having interests within the scope of the standard that are affected by the provisions of the standard have reached substantial agreement on its adoption, (2) formulated in a manner that afforded an opportunity for diverse views to be considered, and (3) designated by the standards body as such a standard for the safe design, manufacture, construction, or operation of nuclear power plants.

- Discussion (Codes and Standards)
  - Preliminary language encourages use of consensus codes and standards as required by the National Technology Transfer and Advancement Act.
  - Recognizes variety of technologies and designs as well as stated desire of some stakeholders to adopt standards outside of typical LWR standards development organizations (e.g., ISO or other international standards).
  - Considering using NRC endorsement of guidance documents versus incorporation of standards into the regulations.
  - Capture of acceptable standards in guidance increases efficiency by avoiding routine rulemakings related to the revision of incorporated standards in the regulations.

Second Iteration – Analysis (PRA)

§ 53.450 Analysis Requirements

(a) *Requirement to have a probabilistic risk assessment*. A probabilistic risk assessment (PRA) of each advanced nuclear plant [reminder – plant definition to include multi-module and multi-source] must be performed to identify potential failures, degradation mechanisms, susceptibility to internal and external hazards, and other contributing factors to unplanned events that might challenge the safety functions identified in § 53.230 **and to support demonstrating that each advanced nuclear plant meets the second tier safety criteria of § 53.220(b).**

- Discussion (PRA)
  - Maintains requirement in Part 53 for PRA consistent with evolution of risk-informed approaches but provide alternatives to PRA for design and analysis processes (paragraph (b)) and to support the licensing and regulatory programs being developed in subsequent subparts
  - Staff is engaged in ongoing discussions on how to ensure the level of effort required for a PRA is commensurate with the complexity of the subject reactor design while also ensuring possible deployment of advanced reactors poses no undue risk to public health and safety.

Past and Present Uses of the PRA

- Identify severe accident vulnerabilities and to provide insights which support the conclusion that the plant design, construction, and operation provides reasonable assurance no undue risk to public health and safety.
- Demonstrate that the plant meets the Commission's safety goals.
- Support the environmental review required by 10 CFR Part 51, specifically, the evaluation of severe accident mitigation design alternatives:
  – RG 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," Rev. 3, September 2018
  – COL-ISG-029, "Environmental Considerations Associated with Micro-reactors," October 28, 2020
- For applications based on the LMP guidance, the PRA is used to select licensing basis events, classify SSCs, and to inform the DiD evaluation.

Past and Present Uses of the PRA (Cont'd)

- For applications not based on the LMP guidance, the PRA may be used to support the process used to demonstrate whether the regulatory treatment of non-safety systems (RTNSS) is sufficient and, if appropriate, identify the SSCs included in RTNSS.
- The results and insights of the PRA are used to identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as:
  - Inspection, testing, analysis, acceptance criteria,
  - TS, and
  - Combined operating license action items and interface requirements.
- The PRA may be used to support various voluntary risk-informed applications (e.g., risk-informed inservice inspection) that may be included in the licensing application.
- The PRA may be used to inform the scope of staff's review; see SRM-COMGBJ-10-0004/COMGEA-10-0001 (ML102510405).
- The results and insights of the PRA are used to support the reactor oversight program.

Searching for Initiating Events (Adapted from the NLWR PRA Standard)

- Identify initiating events that:
  - Challenge normal plant operation (when plant is at-power) or the ability to sustain safe shutdown or low-power conditions (when not at-power), and
  - Require successful mitigation to prevent a release of radioactive material.
- Use a structured, systematic process that accounts for plant- or design-specific features, such as:
  - Master logic diagrams
  - Heat balance fault trees
  - Process hazards analysis
  - Failure modes and effects analysis
- Analyze operating procedures and practices.
- Review existing lists of known initiators applicable to the specific reactor type and design.

Searching for Initiating Events (Cont'd) (Adapted from the NLWR PRA Standard)

- Consider external hazards (e.g., seismic), including initiating events caused by a combination of hazards (e.g., seismically induced fires).
- Review operating experience, including similar plants.
- Perform a systematic evaluation of each system down to the subsystem or train level and including support systems in each modeled plant operating state.
- Include initiating events resulting from multiple failures if the equipment failures result from a common cause.
- Interview resources knowledgeable in plant design or operation.
- Include initiators that impact two or more sources of radioactive material

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Addressing Lack of Operating Experience

<table>
<tr><th>Type of Data/Information</th><th>Methods</th></tr>
<tr><td>Internal initiating event frequencies</td><td rowspan="2">• Many can be estimating using LWR or relevant non-nuclear information
• Bayesian estimation methods
• Formal expert elicitation</td></tr>
<tr><td>Component failure rates</td></tr>
<tr><td>Common-cause failures (CCFs)</td><td>• Use existing CCF models (e.g., alpha factors)
• Use existing generic information derived from LWR experience</td></tr>
<tr><td>Test/maintenance availabilities</td><td>• Use component failure rates
• Controlled by technical specifications (surveillance test intervals and allowed outage times)</td></tr>
<tr><td>Human error probabilities</td><td rowspan="3">• Does not require design-specific operating experience
• Use existing methods</td></tr>
<tr><td>External hazard frequencies</td></tr>
<tr><td>External hazard fragilities</td></tr>
</table>

Addressing Lack of Operating Experience (Cont'd)

- PRA provides a framework for assessing uncertainties:
  - Parametric uncertainties
  - Modeling uncertainties
  - Completeness uncertainties
- PRA helps to put uncertainties into perspective.
  - Which events contribute to the overall uncertainty?
  - Are these events also risk significant?

Second Iteration – Analysis (Use of PRA)

§ 53.450 Analysis Requirements

(b) Requirement to use PRA, <u>other generally accepted risk-informed approach for systematically evaluating engineered systems, or combination thereof</u> to:

- **Determine LBEs**
- **Support safety classification of SSCs**
- **Evaluate defense in depth**

- Discussion (Use of PRA)
  - Change intended to support alternative approaches to a PRA
  - Worded in terms of "generally accepted" to support possible standards or other guidance documents
  - The use of guidance, Part 53 rule language, or revisions to Part 50 are being explored as possible ways to accommodate deterministic approaches for performing design and analysis

Second Iteration – Analysis Requirements (c – g)

§ 53.450 Analysis Requirements
(c) Maintenance and upgrade of analyses
(d) Qualification of analytical codes
(e) Analyses of LBEs (added)
(f) Analysis of DBAs
(g) Other required analyses

- Discussion (Analysis Requirements)
  - Clarification of maintenance and upgrading of analyses (referring to codes and standards)
  - Maintain placeholder for other required analyses to address fire protection, aircraft impact, and specific beyond design basis accidents.

Second Iteration – Analysis Requirements (c – g)

§ 53.450(e) Analyses of licensing basis events [New sub-paragraph]

(e) *Analyses of licensing basis events*. Analyses must be performed for licensing basis events ranging from anticipated operational occurrences to very unlikely event sequences with estimated frequencies well below the frequency of events expected to occur in the life of the advanced nuclear plant. The licensing basis events must be identified using insights from a PRA, other generally accepted risk-informed approach for systematically evaluating engineered systems, or combination thereof to systematically identify and analyze equipment failures and human errors. The analyses must address event sequences from initiation to a defined end state and demonstrate that the functional design criteria required by § 53.420 provide sufficient barriers to the unplanned release of radionuclides to satisfy the second tier safety criteria of § 53.220(b) and provide defense in depth as required by § 53.250.

- Discussion (Analyses of LBEs)
  - Section added to clarify requirements for LBEs, including analysis from initiation to a defined end state
  - Staff considering further clarification for anticipated operational occurrences in terms of acceptance criteria beyond QHOs and defense in depth

Second Iteration – Analysis Requirements (c – g)

§ 53.450 (f) Analysis of design basis accidents

(f) *Analysis of design basis accidents.* The analysis of licensing basis events required by § 53.240 and § 53.450(e) must include analysis of a set of design basis accidents that address possible challenges to the safety functions identified in accordance with § 53.230. Design basis accidents must be selected from those unanticipated event sequences with an upper bound frequency of less than one in 10,000 years as identified using insights from a PRA, other generally accepted risk-informed approach for systematically evaluating engineered systems, or combination thereof to systematically identify and analyze equipment failures and human errors. The events selected as design basis accidents should be those that, if not terminated, have the potential for exceeding the safety criteria in § 53.210(b**). The design-basis accidents selected must be analyzed using deterministic methods that address event sequences from initiation to a <u>safe stable end state</u> and assume only the safety-related SSCs identified in § 53.460 and human actions addressed by § 53.8xx (reference to concept of operations sections of Subpart F) are available to perform the safety functions identified in accordance with § 53.230.** The analysis must conservatively demonstrate compliance with the safety criteria in § 53.210(b).

- Discussion (DBAs)
  - Revised to clarify that analysis is to address sequences from initiation to a safe stable end state.

Second Iteration – Safety Classification

§ 53.460 Safety Categorization and Special Treatment

(a) SSCs and human actions must be classified according to their safety significance. **The categories must include "Safety Related" (SR), "Non-Safety Related but Safety Significant" (NSRSS), and "Non-Safety Significant" (NSS), as defined in subpart A of this part.**

- Discussion
  - Editorial changes to remove material duplicating preliminary rule language in other sections
  - Maintaining for now the specific categories of safety related, non-safety related but safety significant, and non-safety significant

Second Iteration – Analytical Margins and Operating Flexibilities

§ 53.470 Application of Safety Margins to Operational Flexibilities (No Change) Where an applicant or licensee so chooses, design criteria more restrictive than those defined in § 53.220(b) may be adopted to support operational flexibilities (e.g., emergency planning requirements under Subpart F of this part). In such cases, applicants and licensees must ensure that the functional design criteria of § 53.420(b), the analysis requirements of § 53.450, and identification of special treatment of SSCs and human actions under § 53.460 reflect and support the use of alternative design criteria to obtain additional analytical safety margins. Licensees must ensure that measures taken to provide the analytical margins supporting operational flexibilities are incorporated into design features and programmatic controls and are maintained within programs required in other Subparts.

- Discussion
  - No change; Released related requirements in Subpart F to support public meeting on May 6th

Feedback – Design Control Quality Assurance and Design Interfaces

First Iteration
§ 53.480 Design Control Quality Assurance
§ 53.490 Design Interfaces

- Questions/comments on quality assurance and design interfaces
  - Many stakeholders reserving comments pending release of other subparts

- Discussion
  - No change; Released related requirements in Subpart F to support public meeting on May 6th

Feedback – Non-Radiological Hazards

- Non-Radiological Hazards
  - Some ACRS members noted inclusion of non-radiological hazards should be considered in Part 53, such as chemical releases.
    - Staff has this issue under consideration and recognizes existing frameworks for addressing this multi-jurisdictional topic
    - Does ACRS have feedback on this topic that could inform the Staff's ongoing considerations?

Final Discussion and Questions

Acronyms and Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Document Access Management System
AEA	Atomic Energy Act
ALARA	As low as reasonably achievable
ANS	American Nuclear Society
AOO	Anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BDBEs	Beyond design basis events
CCF	Common cause failure
CFR	Code of Federal Regulations
CR	Control rod withdrawal
CT	Circulator trip
DBAs	Design basis accidents
DG	Draft guidance

DiD	Defense in depth
EAB	Exclusion area boundary
EP	Emergency planning
EPA	U.S. Environmental Protection Agency
F-C	Frequency consequence
FMEA	Failure modes and effects analysis
FW	Steam generator feedwater pump trip
HPB	Helium pressure boundary
IAB	Intake air bypass
ISO	International Standards Organization
ITAAC	Inspection, test, analyses, acceptance criteria
LBEs	Licensing basis events
LD	Large helium depressurization
LF	Loss of primary flow

Acronyms and Abbreviations

LMP	Licensing modernization project
LO	Loss of offsite power
LPZ	Low-population zone
LWR	Light-water reactor
MD	Medium helium depressurization
MHTGR	Modular high-temperature gas-cooled reactor
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NLWR	Non-light-water reactor
NRC	U.S. Nuclear Regulatory Commission
NSRSS	Non-safety related but safety significant
NSS	Non-safety significant
PAG	Protective action guide

PC	Plant condition
PPC	Porcelain polycarbonate
PRA	Probabilistic risk assessment
QHO	Quantitative health objective
RAP	Reliability assurance program
Rem	Roentgen equivalent man
ROP	Reactor oversight program
RSF	Required safety function
RT	Reactor trip
RTNSS	Regulatory treatment of non-safety systems
SAR	Safety analysis report
SD	Small helium depressurization
SDO	Standard development organization
SFC	Single-failure criterion

Acronyms and Abbreviations

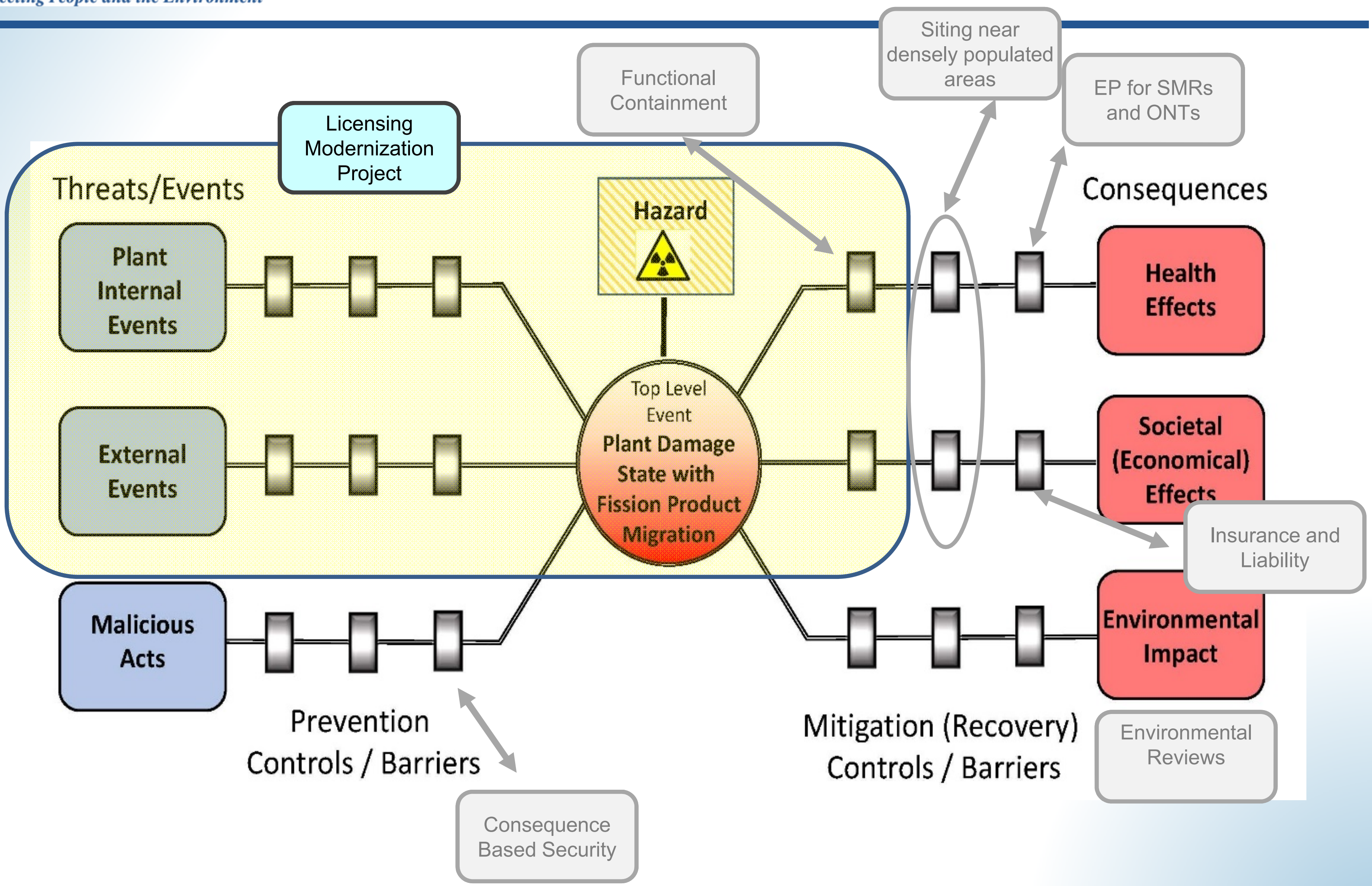
SG	Steam generator rupture
SR	Safety related
SSCs	Structures, systems, components
TS	Technical specifications
TT	Turbine trip

BACKUP SLIDES

Part 53 Rulemaking Schedule

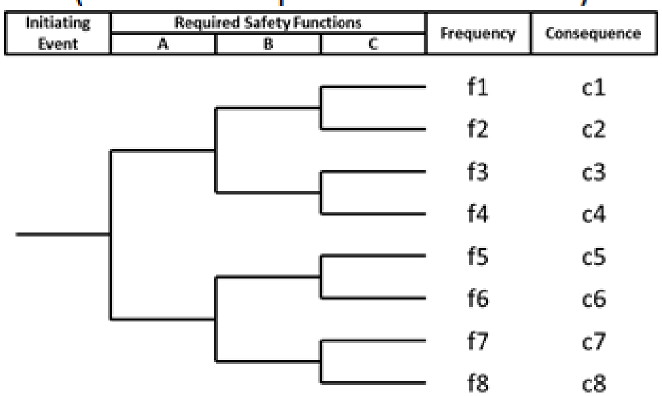
Milestone Schedule	
Major Rulemaking Activities/Milestones	Schedule
Public Outreach, ACRS Interactions and Generation of Proposed Rule Package	Present to April 2022 (11 months)
Submit Draft Proposed Rule Package to Commission	May 2022
Publish Proposed Rule and Draft Key Guidance	October 2022
Public Comment Period – 60 days	November and December 2022
Public Outreach and Generation of Final Rule Package	January 2023 to February 2024 (14 months)
Submit Draft Final Rule Package to Commission	March 2024
Office of Management and Budget and Office of the Federal Register Processing	July 2024 to September 2024
Publish Final Rule and Key Guidance	October 2024

Integrated Approach

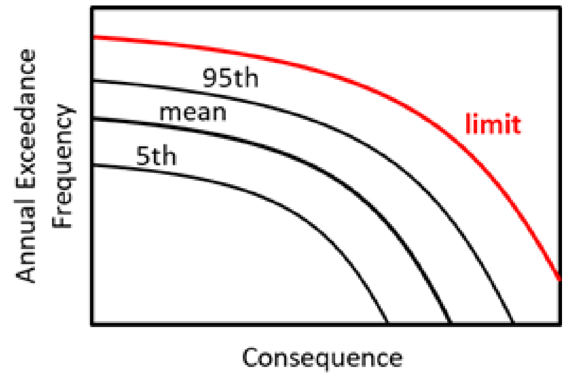


Presenting PRA Results

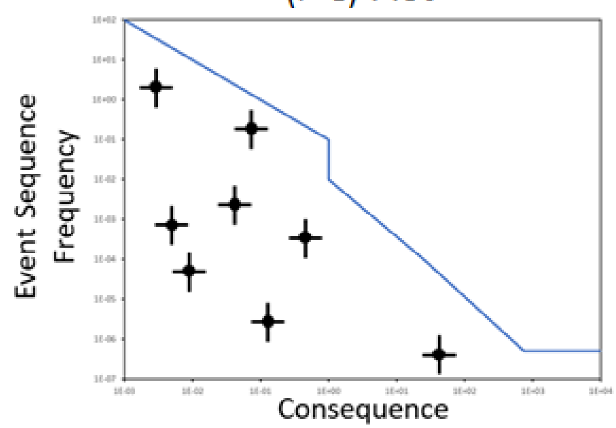
Event Tree
(accident sequence delineation)



Complementary Cumulative Distribution Function (CCDF)



Frequency-Consequence (F-C) Plot



Cumulative Risk

- point estimate = sum of the products of frequency x consequence

$$risk = \sum_i f_i c_i$$

- mean and percentiles

Cumulative Risk Metrics

- QHOs in the Commission's Safety Goal Policy Statement
  - The risk to an average individual in the vicinity of a nuclear power plant [1 mile] of prompt fatalities that might results from reactor accidents should not exceed 0.1% of the sum of prompt fatality risk resulting from other accidents to which members of the U.S. population are generally exposed [5E-7/y].
  - The risk to the population in the area near a nuclear power plant [10 miles] that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes [2E-6/y].
  - Compare mean risks to QHOs, and consider the uncertainties
  - Basis: NUREG-0880, "Safety Goals for Nuclear Power Plant Operation," Rev. 1, ML071770230, May 1983.
- LMP: The total mean frequency of exceeding a site boundary dose of 100 mrem < 1/plant-year (based on 10 CFR 20).

Large Release Frequency (LRF)

- In its safety goal policy statement, the Commission proposed a general performance guideline for further staff examination:
  - The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation
  - Rationale as explained by Forrest Remick (former Director of Office of Policy Evaluation, former ACRS member, and former Commissioner) in a memorandum dated 3/2/1993 (ML051660709) to James Taylor (former EDO):
    - The proposed SGPS included a goal for core-damage frequency (CDF) < 1E-4/y
    - The ACRS wanted to include a goal for conditional containment failure probability (CCFP) < 0.1
    - The LRF goal was developed to break the deadlock between the staff and ACRS
    - (1E-4/y CDF) x (0.1 vessel breach probability) x (0.1 CCFP) = 1E-6 LRF
- In SRM-SECY-89-102 (ML051660712), the Commission made clear that LRF applies to all reactor designs (LWRs and NLWRs).
- As discussed in SECY-93-138, the staff abandoned efforts to anchor LRF to the QHOs (LRF is more conservative).
- There is no NRC definition for LRF; Part 52 applicants have been allowed to propose various definitions.

Large Release vs. Large Early Release

- JCNRM definition of large release (approved 4/2/2021): The release of airborne fission products to the environment such that there are significant off-site impacts. Large release and significant off-site impacts may be defined in terms of quantities of fission products released to the environment, status of fission product barriers and scrubbing, or dose levels at specific distances from the release, depending on the specific analysis objectives and regulatory requirements.

- RG 1.200 implied definition of large early release: A rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there is the potential for early health effects. (Such accidents generally include unscrubbed releases associated with early containment failure shortly after vessel breach, containment bypass events, and loss of containment isolation.)

Core-Damage Frequency (CDF) and Large Early Release Frequency (LERF)

- For large LWRs:

$$LERF < 10^{-5}/y \quad \Rightarrow \quad individual\ prompt\ fatality\ risk < 5 \times 10^{-7}/y$$
$$CDF < 10^{-4} \quad \Rightarrow \quad individual\ latent\ cancer\ fatality\ risk < 2 \times 10^{-6}/y$$

  – Used when developing RG 1.174 (late 1990s)
  – Technical basis documented in NUREG-1860, Appendix D (based on NUREG-1150 results)

- In SRM-SECY-12-0081,the Commission approved the staff's recommendation that new reactors transition from LRF to LERF at or before initial fuel load.

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Protecting People and the Environment

CCDF Representation of Risk

- Used in traditional PRAs (e.g., WASH-1400, NUREG-1150)
- Considered during development of NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing"
  - Deferred - how to establish the acceptance criterion?
  - Discussed in ACRS letter dated September 26, 2007
- Public comment on DG-1353 [RG 1.233] by former ACRS Member Rich Denning and Vinod Mubayi (one of the authors of NUREG-1860) recommended the development of a CCDF criterion in lieu of the frequency-consequence target:
  - Comment: ML19158A457
  - Staff response: ML20091L696
  - Discussed at ACRS Future Plant Design Subcommittee meeting held July 20, 2020

Frequency-Consequence Plot

- Uses include:
  - MHTGR pre-application (1989)
  - NUREG-1860 (2007)
  - NGNP Licensing Strategy (2008)
  - NEI 18-04 (2019)
- In NEI 18-04:
  - The F-C Target is used as a tool to identify risk-significant event sequence families and SSCs
  - **The F-C Target is <u>not</u> an acceptance criterion!**